# Enclosure 2

# License Amendment Request for Callaway Risk-Informed Approach to Resolution of Generic Letter 2004-02

ATTACHMENTS:

2-1 List of Regulatory Commitments

2-2 Technical Specification Page Markups

2-3 Technical Specification Bases Page Markups (for information only)

2-4 Re-typed Technical Specification Pages

2-5 Final Safety Analysis Report Page Markups (for information only)

**License Amendment Request for Callaway Risk-Informed Approach to Resolution of Generic Letter 2004-02**

Subject: Pursuant to 10 CFR 50.90, Union Electric Company (d.b.a. Ameren Missouri) requests amendment of Renewed Operating License NPF-30 for Callaway Plant, Unit 1. The proposed amendment would revise the Final Safety Analysis Report and Technical Specifications for Callaway Plant, Unit 1 using a risk-informed approach to address safety issues discussed in Generic Safety Issue 191, “Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance” [1], and NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors" [18].

## 1. SUMMARY DESCRIPTION

In accordance with 10 CFR 50.59(c)(1)(i) and (c)(2)(viii), Ameren Missouri requests an amendment to Operating License NPF-30 for Callaway Nuclear Plant Unit 1 (Callaway) pursuant to 10 CFR 50.90. The proposed amendment would:

1. Revise the licensing basis as described in the Callaway Final Safety Analysis Report (FSAR) to allow the use of a risk-informed approach to address safety issues discussed in Generic Letter (GL) 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors" [2]. The risk-informed approach is consistent with the guidance of NRC Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" [3].

2. Revise the Technical Specification (TS) for the Emergency Core Cooling System (ECCS) by deleting Surveillance Requirement (SR) 3.5.2.8 in TS 3.5.2, "ECCS - Operating," and deleting its mention from SR 3.5.3.1 in TS 3.5.3, "ECCS - Shutdown."

3. Add new TS 3.6.8, “Containment Sumps,” with appropriate Conditions, Required Actions and Completion Times, including new SR 3.6.8.1 for visual inspection of the containment sumps.

4. Revise TS 5.5.15, “Safety Function Determination Program,” to clarify its application when a supported system is made inoperable by a single TS support system.

In accordance with 10 CFR 50.59(c)(2)(viii), a license amendment shall be obtained prior to implementing a proposed change if the change would result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analysis. Ameren Missouri proposes to amend the Callaway Operating License in order to add risk-informed methodology to FSAR Chapter 6, "Engineered Safety Features." (Chapters 3, "Design of Structures, Components, Equipment, and Systems," and 15, "Accident Analysis," will also refer to it.) Ameren Missouri also proposes to revise the Technical Specifications for the ECCS and Containment Spray System (CSS) in accordance with 10 CFR 50.59(c)(1)(i) and the NRC Safety Evaluation of Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-567-A, Rev. 1 "Add Containment Sump TS to Address GSI-191 Issues" [4]. The proposed TS changes would align the Callaway Technical Specifications with the risk-informed methodology change, and the proposed changes would apply only for the effects of debris as described in GSI-191 and GL 2004-02.

## 2. DETAILED DESCRIPTION

The proposed change associated with the change in methodology is to use a risk-informed approach to determine the design requirements needed to address the effects of loss-of-coolant accident (LOCA) debris instead of a traditional deterministic-only approach. The details of the approach are provided in Enclosure 3 of this submittal. The debris analysis covers a full spectrum of postulated LOCAs, including a range of partial breaks and double-ended guillotine breaks (DEGBs), for all pipe sizes up to and including the design-basis accident (DBA) LOCA in order to provide assurance that the most severe postulated loss-of-coolant accidents are evaluated. The deterministic licensing basis will continue to apply to LOCA break sizes that generate fine fiber debris that is bounded by Callaway plant-specific testing. The proposed methodology change would apply for LOCAs that can generate and transport fine fiber debris that is not bounded by the plant-specific testing. In the risk-informed approach, Callaway conservatively relegates to failure the LOCA break sizes that can generate and transport fine fiber debris that is not bounded by the Callaway plant-specific testing.

The proposed risk-informed methodology change would apply NUREG-1829 [5] to determine the break frequency for the smallest of those breaks to obtain the highest frequency and uses that frequency as the incremental (delta) core damage frequency ($\Delta$CDF) for comparison to the criteria in RG 1.174. The results of the evaluation show that the risk from the proposed change is "very small" in that it is in Region III of RG 1.174. The methodology includes conservatisms in the plant-specific testing and in the assumption that all the unbounded breaks are relegated to failure.

The proposed TS changes associated with the change in methodology would include a new TS, i.e., TS 3.6.8, "Containment Sumps," that contains Conditions and Required Actions with Completion Times that apply when the Limiting Condition for Operation (LCO) is not met. The ECCS and CSS are the only TS systems that depend on the containment sumps as a support system, and are therefore the only systems that are directly subject to the effects of debris. The purpose of the new TS is to have a TS dedicated to the containment sumps in lieu of having the sumps addressed only as a support system behind the ECCS and CSS Technical Specifications, and to establish Conditions and Actions that address the concerns of potential debris effects, including a required action time (Completion Time) for restoring compliance with the LCO that is commensurate with the very low risk associated with debris effects. The proposed action would be required if the limit for analyzed LOCA-generated and transported debris is determined to be exceeded, which is based on the amount of debris used in the Callaway plant-specific testing. The proposed completion time for this TS action is 90 days. In addition, a periodic surveillance to verify by visual inspection that the containment sumps do not show structural damage, abnormal corrosion, or debris blockage would be established per proposed SR 3.6.8.1. A more complete description of the proposed TS changes is provided in Section 2.4.

The proposed change to TS 5.5.15 "Safety Function Determination Program," (SFDP) clarifies its application when a supported system is made inoperable by the inoperability of a single TS support system.

The proposed change to the licensing basis implements a risk-informed approach in lieu of an entirely deterministic method to demonstrate acceptable system responses to a LOCA. In conjunction with the proposed license amendment request (LAR), Callaway is also requesting exemptions from 10 CFR 50.46(a)(1), General Design Criterion (GDC) 35, GDC 38 and GDC 41, as provided in Attachments 1-1 through 1-4 of Enclosure 1 of this submittal. Upon approval of the licensing basis changes, Callaway will make conforming updates to the FSAR. The FSAR markups are attached for the staff's information.

## 2.1 System Design and Operation

The methodology change affects the analysis of systems and functions that are susceptible to the effects of LOCA debris. The affected systems are those that are supported by the strainers and sumps during the recirculation phase of LOCA mitigation, which are the ECCS (i.e., residual heat removal (RHR) pumps, safety injection (SI) pumps, and centrifugal charging pumps (CCPs)) and the CSS. The associated functions and associated regulations are:

- Emergency Core Cooling: 10 CFR 50.46(a)(1) and GDC 35
- Containment Heat Removal: GDC 38
- Containment Atmosphere Cleanup: GDC 41

<u>Emergency Core Cooling System</u>

The ECCS is designed to cool the reactor core and provide shutdown capability subsequent to the following accident conditions:

1. LOCA, including a pipe break or a spurious relief or safety valve opening in the reactor coolant system (RCS) which would result in a discharge larger than that which could be made up by the normal makeup system.

2. Rupture of a control rod drive mechanism, causing a rod cluster control assembly ejection accident.

3. Steam or feedwater system break accident, including a pipe break or a spurious relief or safety valve opening in the secondary steam system which would result in an uncontrolled steam release or a loss of feedwater.

4. A steam generator tube failure.

The primary function of the ECCS is to provide emergency core cooling in the event of a LOCA resulting from a break in the primary RCS or to provide emergency boration in

the event of a steam/or feedwater break accident resulting from a break in the secondary steam system.

The ECCS is safety related and is required to function following a DBA and to achieve and maintain the plant in a safe shutdown condition.

The ECCS meets the following design bases:

1. Except for the refueling water storage tank (RWST), the ECCS is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles (GDC 2). The RWST is designed to seismic Category I requirements only.

2. The ECCS is designed to remain functional after a safe shutdown earthquake (SSE) and to perform its intended function following the postulated hazards of fire, internal missiles, or pipe break (GDC 3 and 4).

3. Safety functions can be performed, assuming a single active component failure coincident with the loss of offsite power (GDC 35).

4. The active components are capable of being tested during plant operation. Provisions are made to allow for in-service inspection of components at appropriate times specified in the ASME Boiler and Pressure Vessel Code, Section XI (GDC 36 and 37).

5. The ECCS is designed and fabricated to codes consistent with the quality group classification assigned by RG 1.26 and the seismic category assigned by RG 1.29. The power supply and control functions are in accordance with RG 1.32.

6. The capability to isolate components or piping is provided so that the ECCS safety function will not be compromised. This includes isolation of components to deal with leakage or malfunctions and to isolate nonsafety-related portions of the system (GDC 35).

7. The containment isolation valves in the system are selected, tested, and located in accordance with the requirements of GDC 54 and 55 and 10 CFR 50, Appendix J, Type A testing.

8. ECCS equipment design ensures acceptable performance for all environments anticipated under normal, testing, and design-basis accident conditions.

9. The functional requirements of the ECCS are derived from 10 CFR 50, Appendix K limits for fuel cladding temperature, etc., following any of the above accidents, as delineated in 10 CFR 50.46. The subsystem functional parameters are

integrated so that the Appendix K requirements are met over the range of anticipated accidents and single failure assumptions.

There are no power generation design bases for the ECCS function. Portions of the ECCS are also portions of the residual heat removal system (RHRS) and chemical and volume control system (CVCS), and are used during normal power operation.

10 CFR 50.46(b) provides the following criteria to judge the adequacy of the ECCS.

1. Peak clad temperature calculated shall not exceed 2,200°F.

2. The calculated total oxidation of the clad shall nowhere exceed 0.17 times the total clad thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the clad with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the clad cylinders surrounding the fuel, excluding the clad around the plenum volume, were to react.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by long-lived radioactivity remaining in the core.

In addition to, and as an extension of the 10 CFR 50.46(b) Final Acceptance Criteria, two accidents have more specific criteria, as described below.

In the case of the inadvertent opening of a steam generator relief or safety valve, an additional criterion for adequacy of the ECCS is: Assuming a stuck rod cluster control assembly (RCCA), offsite power available, and a single failure in the engineered safety features, there will be no return to criticality after reactor trip for a steam release equivalent to the spurious opening (with failure to close), of the larger of a single steam dump, relief, or safety valve.

For a steam system piping failure, the added criterion is: Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safety features, the core remains in place and intact.

These two accidents and others listed in the ECCS description (e.g., main steam line break, reactor coolant pump seal leak) do not require the use of the containment sump strainers to mitigate accident generated and transported debris. Enclosure 3, the technical analysis performed for this LAR, provides further details on how these secondary risk contributors are assessed.

Containment Heat Removal System (CHRS)

The functional performance objective of the containment heat removal system, as an engineered safety features system, is to reduce the containment temperature and pressure following a LOCA or main steam line break (MSLB) accident by removing thermal energy from the containment atmosphere. These cooling systems also serve to limit offsite radiation levels by reducing the pressure differential between the containment atmosphere and the external environment, thereby diminishing the driving force for the leakage of fission products from the containment to the environment. The containment heat removal systems include the residual heat removal system (see ECCS system description), the CSS, and the containment cooling system (CtCS).

The CSS:

The CSS consists of two separate trains of equal capacity, each independently capable of meeting the design bases. Each train includes a containment spray pump, spray header and nozzles, spray recirculation path, valves, and the necessary piping, instrumentation, flushing connections, and controls.

The RWST supplies borated injection water to the CSS. Each train takes suction from separate containment sumps during the recirculation phase.

The CSS provides a spray of cold or subcooled borated water from the upper regions of the containment to reduce the containment pressure and temperature during either a LOCA or MSLB inside the containment.

Each CSS pump discharges into the containment atmosphere through an independent spray header. The spray headers are located in the upper part of the reactor building to allow maximum time for the falling spray droplets to reach thermal equilibrium with the steam-air atmosphere. The condensation of the steam by the falling spray results in a reduction in containment pressure and temperature. Each spray train provides adequate coverage to meet the design requirements with respect to both containment heat removal and iodine removal.

In the CSS, only the containment sumps, the trisodium phosphate baskets, the spray headers, nozzles, and associated piping and valves are located within the containment. The remainder of the system is located within the auxiliary building, separated from that portion in the containment by motor-operated isolation valves.

Following a large break LOCA, the containment spray during the injection phase will be a boric acid solution having a pH of about 4.5. The desired pH level is greater than 7.0 to assure iodine retention in the sumps, to limit corrosion and the associated production of hydrogen, and to limit chloride induced stress-corrosion cracking of austenitic stainless steels. To adjust the sump solution pH into the desired range, a minimum of 9,000 pounds of trisodium phosphate dodecahydrate ($NA_3PO_4 \cdot 12H_2O \cdot 1/4\ NaOH$) is stored in two baskets, one adjacent to each containment sump, at an elevation to

assure dissolution after a LOCA. This amount of trisodium phosphate is sufficient to assure that the equilibrium sump solution pH will be greater than or equal to 7.1.

The CSS meets the following design bases:

1. The CSS is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, or external missiles (GDC 2).

2. The CSS is designed to remain functional after a SSE or to perform its intended function following the postulated hazard of a pipe break (GDC 3 and 4).

3. Safety functions can be performed, assuming a single active component failure coincident with the loss of offsite power (GDC 38).

4. The active components are capable of being tested during plant operation. Provisions are made to allow for in-service inspection of components at appropriate times specified in the ASME Boiler and Pressure Vessel Code, Section XI (GDC 39 and 40).

5. The CSS is designed and fabricated to codes consistent with the quality group classification assigned by Regulatory Guide 1.26 and the seismic category assigned by Regulatory Guide 1.29. The power supply and control functions are in accordance with Regulatory Guide 1.32.

6. The capability of isolating components or piping is provided so that the CSS safety function will not be compromised. This includes isolation of components to deal with leakage or malfunctions (GDC 38).

7. The containment isolation valves in the system are selected, tested, and located in accordance with the requirements of GDC 54 and 56 and 10 CFR 50, Appendix J, Type A testing.

8. The CSS, in conjunction with the containment fan cooler system and the emergency core cooling system, is designed to be capable of removing sufficient heat and subsequent decay heat from the containment atmosphere following the hypothesized LOCA or MSLB to maintain the containment pressure below the containment design pressure.

9. The CSS remains operable in the accident environment.

10. The containment spray water does not contain substances which would be unstable in the thermal or radiolytic environment of the LOCA or cause extensive corrosive attack on equipment.

11. The CSS is designed so that adequate net positive suction head (NPSH) exists at the suction of the containment spray pumps during all operating phases, in accordance with Regulatory Guide 1.1.

12. The CSS is designed to prevent debris which could impair the performance of the containment spray pumps, valves, eductors, or spray nozzles from entering the recirculation piping. Design is in accordance with Regulatory Guide 1.82.

The CtCS:

The CtCS, in conjunction with the containment HVAC systems, functions during normal plant operation to maintain a suitable atmosphere for equipment located within the containment. Subsequent to a DBA within the containment, the containment cooling system provides a means of cooling the containment atmosphere to reduce pressure and thus reduce the potential for containment leakage of airborne and gaseous radioactivity to the environment.

The CtCS provides cooling by recirculation of the containment air across air-to-water heat exchangers. The bulk of this cooled air is supplied to the lower regions of the steam generator compartments. The remaining air is supplied to the instrument tunnel and at each level (operating floor and below) of the containment outside the secondary shield wall. The air supplied to each steam generator compartment is drawn upwards through the compartments by the hydrogen mixing fans and discharged into the upper elevations of the containment.

Combustible Gas Control in Containment

The hydrogen control system (HCS) is an engineered safety feature which serves to control combustible gas concentrations in the containment. The HCS consists of redundant hydrogen recombiners, a redundant hydrogen mixing system, redundant hydrogen monitoring subsystem, and a backup hydrogen purge subsystem.

Sump Design

To address debris-related concerns associated with GSI-191 and in response to the debris issues identified in GL 2004-02, new containment sump strainers were installed in April 2007 to replace the previously installed screens. The wetted surface area of the strainers was increased from approximately 400 square feet to approximately 6,600 square feet. The screen-hole size of the strainers was reduced from 1/8 inches to 0.045 inches. Small particles in water entering the suction pipe from the sump cannot clog the containment spray nozzles (which have a minimum constriction size of 7/16”). Previously, the sump screens had extended above the maximum, post-accident containment water level and would not be submerged. The replacement sump strainers are now inside the containment sump pit ensuring maximum strainer surface area is available during post-accident recirculation mode. Installation of the new strainers did

not affect the independence and redundancy of the sumps; each one of the two sumps has sufficient capacity to serve one of the redundant halves of the ECCS and CSS.

The sump strainer design implemented by these modifications meets the current design-basis requirements with respect to NPSH and ECCS performance. The sumps are designed according to RG 1.82 Revision 0, dated June 1974, which recommends a calculation of sump screen head loss due to 50% debris blockage of the wetted surface. Utilizing the current licensing basis methodology, the pump NPSH is sufficient to accommodate this head loss. The Callaway sumps meet the function to preclude passage of debris particles large enough to damage downstream components in the ECCS and CSS.

The sump strainer design has been evaluated to meet the current licensing basis assumptions for analyzing the effects of post-accident debris blockage and for compliance with 10 CFR 50.46 for long term cooling, GDC 35 for emergency core cooling, GDC 36 for inspection of ECCS, GDC 38 for containment heat removal, GDC 39 for inspection of containment heat removal system, and GDC 41 for containment atmosphere cleanup.

The containment sumps meet each position of Regulatory Guide 1.82, Revision 0:

1. Two sumps are provided, and each has sufficient capacity to serve one of the redundant trains of the ECCS and CS systems.

2. The redundant sumps are physically separated from each other and from high energy piping.

3. The sumps are located at the 2000' elevation, which is the lowest floor elevation in the reactor building, exclusive of the reactor cavity. The strainers are installed in the recirculation sump pit and extend approximately one foot above the 2000’ elevation of the Reactor Building. The intent is met.

4. The floor is level in the vicinity of the sump. However, a 6-inch concrete curb is provided which prevents high density particles from entering the sump.

5. All drains in the upper regions of the reactor building are terminated in such a manner that direct streams of water which may contain entrained debris will not impinge on the filter assemblies.

6. The containment sump strainers are fabricated from stainless steel perforated plate, including structural reinforcement, and are sufficiently rigid to preclude the use of a trash rack. The structural evaluation for the strainers concludes that the strainers meet the acceptance criteria for all applicable loadings during the recirculation phase of an event. The sumps and strainers are outside the secondary shield wall which provides protection from missiles and large debris. The intent is met.

7. The containment sump strainers are composed of stainless steel perforated plate with 0.045-inch diameter holes. The approach velocity of the recirculation coolant flow at the sump strainer face is less than 0.2 ft/sec.

8. A concrete slab over the containment sump strainers is provided. The containment recirculation sump strainers will be fully submerged following a large break LOCA.

9. The containment sump strainers are designed as seismic Category I and have been evaluated acceptably for all applicable loadings.

10. The containment sump strainers have a nominal 0.045-inch hole size, which precludes particles larger than 0.045 inches from passing through the strainers. The containment spray pump is designed to pass particles less than 1/4 inch in size, and the minimum restriction in the spray system is the 7/16-inch orifice in the spray nozzle.

11. The pump intake location in the sump is horizontal to limit any degrading effects due to vortexing.

12. The containment sump strainers are fabricated from stainless steel. Stainless steel has a low sensitivity to corrosion during power operation and after an event.

13. The containment sump strainers are provided with provisions to allow inspection of the strainer structure and areas downstream of the strainer.

14. In-service inspection requirements consist of visual examination during every scheduled refueling downtime.

The locations of the strainers provide significant protection from dynamic effects such as pipe whip, jet impingement, and missile impacts associated with a high-energy line break. The containment sump strainers are outside the secondary shield wall and are located inside a pit where approximately 1 foot of the strainers is above the reactor containment building floor. A concrete structure is also approximately 7 feet above the strainers, and a concrete wall divides the strainer trains. Structural steel that stabilizes the top of the strainer stacks also provides protection.

The replacement strainer design is a safety improvement that contributed to meeting the RG 1.174 criteria for Region III, "Very Small Changes," for the results obtained from the risk-informed methodology.

## 2.2 Current Technical Specification Requirements

Under the current TS, the operability of the ECCS Subsystems is assured by the capability of the containment sump strainers to limit entry of debris into the sumps and recirculating lines and also allow adequate flow into the system. This capability ensures that the flow and net positive suction head requirements of ECCS are satisfied.

Assurance that containment debris will not block the sump strainers and render the ECCS Subsystem inoperable on emergency recirculation during design-basis accidents is provided by inspection and engineering evaluation.

### 2.2.1 TS 3.5.2 ECCS - Operating

TS 3.5.2 requires two ECCS trains to be OPERABLE during Modes 1, 2, and 3. (Each train must include an ECCS centrifugal charging pump (high-head pump), a safety injection pump (intermediate-head pump), and a residual heat removal pump (low-head pump). The TS provides an Action (i.e., Required Action A.1) that allows 72 hours to restore one or more inoperable trains to meet the LCO requirement of having two ECCS trains OPERABLE, provided at least 100% of the ECCS flow equivalent to a single operable ECCS train is available. (The latter is met by having a combination of one high-head, one intermediate-head, and one low-head pump available from either train.) Reliability analysis has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours. If the inoperable train is not restored within the 72-hour Completion Time, Required Action B.1 allows 6 hours in to be MODE 3, and in parallel, Required Action B.2 allows 12 hours to be in MODE 4.

### 2.2.2 TS 3.5.3 ECCS - Shutdown

TS 3.5.3 requires one ECCS train to be OPERABLE during Mode 4. (For Mode 4, the required OPERABLE train must include an ECCS centrifugal charging pump and a residual heat removal pump.) The TS provides an Action (i.e., Required Action A.1) to initiate action to restore the required ECCS RHR subsystem to OPERABLE status immediately, and an Action (i.e., Required Action B.1) to restore the required ECCS Centrifugal Charging Pump subsystem to OPERABLE status within 1 hour. If at least one ECCS Centrifugal Charging Pump subsystem is not restored to OPERABLE status within the 1-hour Completion Time, Required Action C.1 allows 24 hours to be MODE 5.

### 2.2.3 TS 3.6.6 Containment Spray and Cooling Systems

The TS provides an Action (i.e., Required Action A.1) that allows 72 hours to restore an inoperable containment spray train to OPERABLE status (and 10 days from discovery of failure to meet the LCO) in order to meet the requirement of having two containment spray trains and two containment cooling trains OPERABLE. The 72-hour Completion Time is based on the ability of the remaining operable train to provide adequate coverage to meet design requirements with respect to both containment heat removal and iodine removal. The 72 hours takes into account reasonable time for repairs and the low probability of a DBA occurring during this period. If the inoperable train is not restored within the 72-hour Completion Time, Required Action B.1 allows 6 hours to be in MODE 3, and in parallel, Required Action B.2 allows 84 hours to be in MODE 5. If one containment cooling train is inoperable, the TS provides an Action (i.e., Required Action C.1) that allows 7 days to restore containment spray train to OPERABLE status (and 10 days from discovery of failure to meet the LCO). If the inoperable containment cooling train is not restored within the 7-day Completion Time, Required Action D.1

allows 6 hours to be in MODE 3, and in parallel, Required Action D.2 allows 36 hours to be in MODE 5. If two spray trains or two cooling trains are inoperable, Required Action E.1 allows 6 hours to be in Mode 3, and in parallel, Required Action E.2 allows 36 hours to be in Mode 5.

### 2.2.4 TS 3.6.8 Containment Sumps

There is currently no Containment Sumps TS.

### 2.2.5 TS 5.5.15 Safety Function Determination Program (SFDP)

The current SFDP ensures loss of safety function is detected and appropriate actions are taken. Additionally, other actions may be taken as a result of support system inoperability and corresponding exception to entering supported system Conditions and Required Actions.

## 2.3 Reason for the Proposed Changes

Callaway is proposing to change the Technical Specifications related to the ECCS and CSS in order to address the effects of HELB generated and transported debris. The ECCS and CSS are the only systems potentially affected by debris effects because they are the only systems that are supported by the containment sumps and their strainers.

As noted previously, the purpose of the proposed, new Technical Specification for the containment sumps, i.e., TS 3.6.8, is to have a TS dedicated to the containment sumps in lieu of having the sumps addressed only as a support system behind the ECCS and CSS Technical Specifications, and to establish Conditions and Actions that address the concerns of potential debris effects.

## 2.4 Description of Proposed Changes

Revising the plant's licensing basis to reflect resolution of the concerns identified per GSI-191 and GL 2004-02 requires changes to be made to the Technical Specifications and their associated TS Bases, as well as the FSAR.

### 2.4.1 Proposed TS Changes

The changes proposed for the Callaway Technical Specifications are based on the changes proposed and described in NRC-approved TSTF-567-A, Rev. 1.

Of the various options pursued by licensees for resolving the GSI-191 and GL 2004-02 concerns, it is noted in the approved TSTF-567 package that the TS changes proposed per the TSTF may be considered applicable to all plants, regardless of the GSI-191 closure option selected. In addition, a model license amendment application for adopting the TSTF is included in the TSTF-567 package. In regard to that, however, it is noted in the TSTF package that the model application is intended for use by Option 1

plants and Option 2a plants that have closed, or will close, GSI-191 utilizing an NRC-approved deterministic approach. Option 2b or Option 3 plants using a risk-informed approach are required to submit a plant-specific license amendment request to close GSI-191, and will not use the model application. The proposed TS changes and associated justification can be used by a risk-informed option plant and included as part of the license amendment request with the necessary technical justification.

For the risk-informed Option 2b approach taken for Callaway, Ameren Missouri has determined that the TS changes proposed per TSTF-567 are applicable to Callaway and may be proposed for Callaway as-is. That is, although Callaway's approach to resolving the GSI-191 and GL 2004-02 concerns involves deterministic and risk-informed aspects, and although both of those aspects contributed to establishing the "containment accident generated and transported debris" limits referred to in the new containment sump TS proposed for Callaway (described further below), the combined approach does not require any deviations from the TS changes prescribed by TSTF-567-A (Rev. 1).

Consistent with TSTF-567, the primary change to be made to the Callaway Technical Specifications is a new Technical Specification dedicated to the containment sumps. As previously noted, the sumps are a support system for the ECCS and CSS. In the current Technical Specifications, there is a Surveillance Requirement (SR) for the sumps within the ECCS Technical Specifications, but there is no dedicated Limiting Condition for Operation (LCO) for the sumps. Per the proposed TS changes, the sumps will have their own LCO and SRs under a new TS dedicated to the sumps. This change and some ancillary TS changes, including removal of the sump-related SRs under the ECCS Technical Specifications, as well as a change to the Administrative Controls TS that describes the Safety Function Determination Program, are described in greater detail in the following subsections.

#### 2.4.1.1 Proposed Change to TS 3.5.2, "ECCS – Operating"

Per the current Callaway Technical Specifications, TS 3.5.2, "ECCS – Operating," contains an SR, i.e., SR 3.5.2.8, which requires periodic verification by visual inspection that "each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion." With the creation of a new TS dedicated to the containment sumps (i.e., new TS 3.6.8, as further described in subsection 2.4.1.3 below), SR 3.5.2.8 is no longer needed in its current place since the SRs pertinent to the containment sumps will be specified under the new containment sump TS. As explained in Section 2.4.1.3, the new SR under TS 3.6.8 is broader in scope such that SR 3.5.2.8 is fully included in the new SR.

A corresponding change to the TS Bases for TS 3.5.2 will be made to reflect the removal of SR 3.5.2.8. In accordance with 10 CFR 50.36, changes to the TS Bases will be made in accordance with the Technical Specifications Bases Control Program

following approval of the requested amendment. The TS Bases changes are provided for information only and approval of the Bases is not requested.

### 2.4.1.2 Proposed Change to TS 3.5.3, "ECCS – Shutdown"

TS 3.5.3, "ECCS – Shutdown," contains a Surveillance Requirements section in which one SR is identified, i.e., SR 3.5.3.1. However, this SR identifies a number of applicable surveillances by identifying those SRs in TS 3.5.2 that also apply to SR 3.5.3.1. Thus, by its list of referenced SRs, SR 3.5.3.1 is satisfied by the satisfactory performance of SRs 3.5.2.1, 3.5.2.3, 3.5.2.4, 3.5.2.7, and 3.5.2.8. Since the list of referenced SRs includes SR 3.5.2.8, and since SR 3.5.2.8 is being eliminated from its present location, the reference to SR 3.5.2.8 should be eliminated from SR 3.5.3.1. Further, there is no need for SR 3.5.3 to refer the new/relocated SR under the new containment sump since that SR is now associated with that TS and is no longer associated with the ECCS TS. (That is, for the same reason that SR 3.5.2.8 is being eliminated from TS 3.5.2 as explained above, it should be eliminated from TS 3.5.3 as well.) Summarily, the change to TS 3.5.3 is simply the elimination of "SR 3.5.2.8" from list of SRs contained in SR 3.5.3.1. (A corresponding change to the Bases for TS 3.5.3 is also proposed, as described in Attachment 2-3 to this Enclosure.)

### 2.4.1.3 Proposed New TS 3.6.8, "Containment Sumps"

The principal change proposed for the Callaway Technical Specifications is the addition of an entirely new TS dedicated to the containment sumps, i.e., TS 3.6.8, "Containment Sumps," complete with an LCO and Applicability, Conditions and Required Actions, including specified Completion Times, as well as an applicable SR. (A new Bases section dedicated to this TS is also proposed, as described in Attachment 2-3 to this Enclosure.)

Consistent with the redundant sumps designed for Callaway, LCO 3.6.8 (as proposed) states that "two containment sumps shall be OPERABLE." In regard to its Applicability, it is proposed that Modes 1, 2, 3 and 4 be identified as the applicable Modes. This is consistent the TS requirements for the ECCS and CSS to be Operable in Modes 1, 2, 3 and 4 and the requirement for the containment sumps to be Operable to support those functions/systems. (In Modes 1, 2, and 3, the plant is at normal operating pressure and temperature where generation of design-basis quantities of debris can reasonably be postulated. For Mode 4, there is less energy in the reactor coolant system (RCS) and reduced capability to generate the zones of influence associated with pipe breaks, but Mode 4 is included in the Applicability of TS 3.6.8 nonetheless.) In Modes 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these Modes. Thus, the containment sumps are not required to be Operable in Modes 5 or 6.

For the proposed Actions section of TS 3.6.8, it is necessary to address the condition of having one or more containment sumps inoperable. Consistent with how the Conditions are to be specified per TSTF-567, however, Condition A would address the condition of

having one or more sumps inoperable due solely to a debris issue, and Condition B would address the condition of having one or more sumps inoperable for reasons other than just a debris issue.

Accordingly, proposed Condition A specifies that if one or more containment sumps are "inoperable due to containment accident generated and transported debris exceeding the analyzed limits," then Required Actions A.1, A.2 and A.3 are to be entered. Respectively, these Required Actions require station personnel to: (1) immediately initiate action to reduce containment accident generated and transported debris, (2) perform SR 3.4.13.1 once per 24 hours, and (3) restore the containment sumps to OPERABLE status within 90 days. (Surveillance Requirement 3.4.13.1 requires verification that RCS operational leakage is within limits by performance of an RCS water inventory balance.)

The phrase "containment accident generated and transported debris" is taken directly from TSTF-567 (Rev. 1) and is further defined in the proposed Bases for TS 3.6.8. (See Attachment 2-3 to this Enclosure.) On the basis of that wording, Condition A is meant to apply only for the effects of LOCA generated and transported debris that exceeds the amount that has been analyzed. It does not apply to non-conforming or degraded conditions that are not associated with the LOCA-generated and transported debris, such as a strainer obstructed by a tarp (a condition that makes the strainer non-functional regardless of debris) or a strainer with large gaps that would pass debris fragments which would make non-applicable the risk evaluation based on fine fiber debris. The TS Bases address the analyzed debris amounts/limits applicable to Condition A and Required Action A.1.

The Bases for proposed new TS 3.6.8 (as further described in Attachment 2-3 to this Enclosure) give examples of actions that may be taken to satisfy Required Action A.1, i.e., to "reduce containment accident generated and transported debris." These include the following:

- Removing the debris source from containment or preventing the debris from being transported to the containment sumps;
- Evaluating the debris source against the assumptions in the analysis;
- Deferring maintenance that would affect availability of the affected systems and other LOCA-mitigating equipment;
- Deferring maintenance that would affect availability of primary defense-in-depth systems, such as containment coolers;
- Briefing operators on LOCA debris management actions; or
- Applying an alternative method to establish new limits.

The 90-day Completion Time proposed for Required Action A.3, i.e., for restoring the affected containment sump to Operable status, is the same as what is specified in TSTF-567 (Rev. 1) for this Required Action, as approved by the NRC (and reflected in Consolidated Line Item Improvement Process for the TSTF). The 90-day Completion

Time is reasonable for emergent conditions that involve debris which could be generated and transported under LOCA conditions. The likelihood of an initiating event in the 90-day Completion Time is very small (~1/4 of the LOCA annual frequency). There are margins in the debris generation and transport analyses and in the downstream and in-core effects analyses. 90 days is a reasonable time to restore the affected subsystems to Operable status by completing the identification and implementation of mitigating or compensatory actions, such as removing the debris, securing or containing the debris so that it is not transportable, performing additional analysis to demonstrate Operability, or to obtain regulatory relief (e.g., Enforcement Discretion and/or an Emergency or Exigent TS change). Additionally, the 90-day Completion Time provides clarity for the operators with regard to application of the TS for degraded or nonconforming conditions associated with the effects of LOCA debris.

Proposed Condition B addresses having one or more containment sumps inoperable for reasons other than Condition A. For this Condition, Required Action B.1 must be entered, which requires restoring the containment sump(s) to operable status within the specified Completion Time of 72 hours. As proposed, Required Action B.1 is modified by two notes. Note 1 directs entering the "applicable Conditions and Required Actions of LCO 3.5.2, “ECCS - Operating,” and LCO 3.5.3, “ECCS - Shutdown,” for emergency core cooling trains made inoperable by the containment sump(s)." Note 2 directs entering the "applicable Conditions and Required Actions of LCO 3.6.6, “Containment Spray and Cooling Systems,” for containment spray trains made inoperable by the containment sump(s)."

Proposed Condition C is to be entered if the Required Actions and associated Completion Times for Condition A or Condition B are not met. Per Required Actions C.1 and C.2, as proposed, the plant must be in Mode 3 in 6 hours and Mode 5 in 36 hours, respectively.

One SR is proposed for new TS 3.6.8. Specifically, new SR 3.6.8.1 would require verifying, "by visual inspection, [that] the containment sumps do not show structural damage, abnormal degradation, or debris blockage" (based on the wording prescribed in TSTF-567). The frequency for performance of this SR would be as specified per Callaway's Surveillance Frequency Control Program, which was established per License Amendment 202 of the Callaway Operating License and is described in Administrative Control TS 5.5.18.

It was noted previously that SR 3.5.2.8 is being eliminated in light of the fact that new SR 3.6.8.1 provides the appropriate surveillance for the containment sumps and encompasses the scope of SR 3.5.2.8. This can be discerned by comparing the wording of the former, "Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion," which focuses on the suction inlets and inlet strainers, to the latter, "Verify, by visual inspection, the containment sumps do not

show structural damage, abnormal corrosion, or debris blockage," which focuses on the containment sumps overall.

In regard to this change, therefore, the NRC concluded in their evaluation of TSTF-567, Rev. 1, that "the proposed change is acceptable since the existing requirements are either unchanged or expanded and continue to ensure the containment sump is unrestricted (i.e., unobstructed) and stays in proper operating condition. The proposed SR meets the requirements of 10 CFR 50.36(c)(3) because it provides an SR to assure the necessary quality of systems and components are maintained, that facility operation will be within safety limits, and that the LCOs will be met."

#### 2.4.1.4 Proposed Change to TS 5.5.15, "Safety Function Determination Program (SFDP)"

Ameren Missouri proposes to add the following sentence at the end of TS 5.5.15, "Safety Function Determination Program (SFDP)," to clarify the SFDP, consistent with TSTF-567, Revision 1, and the associated NRC SE:

> When a loss of safety function is caused by the inoperability of a single TS support system, the appropriate Conditions and Required Actions to enter are those of the support system.

The SFDP described in TS 5.5.15 is necessary for the proper implementation of LCO 3.0.6, in regard to addressing certain inoperable equipment conditions when the LCOs of systems involving support-supported system relationships are involved. Specifically, per LCO 3.0.6, when a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with the supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This provision is an exception to LCO 3.0.2 for the supported system. In such cases, LCO 3.0.6 requires an evaluation to be performed in accordance with TS 5.5.15 (the SFDP) to determine whether a loss of safety function exists. If a loss of safety function is determined to exist by the SFDP, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

Thus, for example, if one train of a support system having redundant trains is declared inoperable, cross-train checks are performed to identify whether a loss of safety function exists for the associated, supported system(s) that might also have redundant trains. The cross train check verifies that the supported systems of the redundant support system are Operable, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

Callaway's containment sump design includes intended redundancy such that there are two containment sumps. However, in light of the concerns raised by GSI-191 and GL

2004-02, the two sumps must be considered part of a single support system because containment accident generated and transported debris that could render one sump inoperable could render both sumps inoperable. Declaring one or both sumps inoperable per Condition A of proposed TS 3.6.8 (for one or more containment sumps "inoperable due to containment accident generated and transported debris exceeding the analyzed limits"), and then applying the provisions of the SFDP as currently described in TS 5.5.15, could (unnecessarily) result in declaring both ECCS trains and both CSS trains inoperable such that the Conditions and Required Actions for both trains inoperable would have to be entered under TS 3.5.2 and TS 3.6.6.

By considering the containment sumps to be a single TS support system, the words to be added to TS 5.5.15 would make it clear that when a loss of safety function is caused by the inoperability of a single TS support system, the appropriate Conditions and Required Actions to be entered are those of the support system (in lieu of the supported system).

As noted in the NRC's Safety Evaluation (SE) for approval of TSTF-567 (Rev. 1), the proposed change to the TS-described SFDP is applicable to plants that have more than one containment sump. Further, it is stated in the NRC's SE that the proposed addition to TS 5.5.15 clarifies the intent of the allowance (not to enter the Conditions and Required Actions of the supported systems) provided by LCO 3.0.6 and the SFDP for single-train support systems. It is noted that the proposed change is acceptable since the actions for the support system LCO adequately address the inoperability of that system. Therefore, as required by 10 CFR 50.36(c)(5), the TS-described SFDP (as revised by the proposed change) continues to provide adequate administrative controls to assure safe operation.

### 2.4.2 Proposed FSAR Changes

Upon approval of the licensing basis changes, Ameren Missouri will make the following changes to the Callaway FSAR:

- Add Appendix 6.3A, "Risk-Informed Approach to Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents." This appendix describes the evaluations performed using a risk-informed approach to address GSI-191 concerns including the effects on long-term cooling due to debris accumulation on containment sump strainers for ECCS and CSS in recirculation mode, as well as core flow blockage due to in-vessel effects, following LOCAs This section summarizes the methodology change(s) for which NRC approval is sought.
    - Callaway requests NRC approval of Section 2.0 of Appendix 6.3A since it includes criteria for identifying changes that would require prior NRC approval.
- Make conforming changes to FSAR Table 1.6-2

- Make conforming changes to FSAR Chapter 3 descriptions of evaluations against GDC 35, GDC 38 and GDC 41
- Make conforming changes to FSAR Chapter 6
- Make conforming changes to FSAR Chapter 15

The proposed FSAR changes are provided as mark-ups in Attachment 2.5 to this Enclosure.

## 3. TECHNICAL EVALUATION

### 3.1 Background

GSI-191 concerns the possibility that debris generated during a LOCA could clog the containment sump strainers in pressurized-water reactors (PWRs) and result in loss of NPSH for the ECCS and CSS pumps, thus impeding the flow of water from the sumps. GL 2004-02 requested licensees to address GSI-191 issues, with a focus on demonstrating compliance with the ECCS acceptance criteria in 10 CFR 50.46. GL 2004-02 also requested licensees to perform new, more realistic analyses using an NRC-approved methodology and to confirm the functionality of the ECCS and CSS during design-basis accidents that require containment sump recirculation.

The Callaway risk-informed approach maintains the defense-in-depth measures in place to mitigate the residual risk of strainer or in-vessel issues to address GL 2004-02. These measures include those implemented in response to NRC Bulletin 2003-01 [6] and GL 2004-02 to address the potential for sump strainer clogging and other concerns associated with GSI-191. Additional measures, such as operating procedures and instrumentation to monitor core level and temperature, and actions taken by operators if core blockage is indicated, have been implemented. These actions are not the subject of this license amendment request. Detailed discussion regarding defense-in-depth is provided in Enclosure 3, Attachment 3-4 to this letter. These measures are part of the defense-in-depth for Callaway and remain in place.

The Commission issued Staff Requirements Memorandum (SRM)-SECY-10-0113, "Closure Options for Generic Safety Issue 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance," [7] directing the staff to consider alternative options for resolving GSI-191 and GL 2004-02 that are innovative and creative, as well as risk-informed and safety conscious. Subsequently, through interactions with the staff, South Texas Project Nuclear Operating Company (STPNOC) developed a risk-informed approach to address GSI-191 based on the guidance in RG 1.174, which serves as a pilot for other licensees to adopt. Ameren Missouri has chosen to use this approach as described in the following two sections.

Plant-Specific Testing

Callaway conducted successful plant-specific testing in July 2016 using approved prototypical debris, conservative chemical effects, prototypical simulation of strainer approach flow conditions, and a Callaway strainer module. This plant-specific test is described in more detail in Enclosure 3 and forms the basis for the deterministic scope of the proposed methodology change.

Use of a Risk-Informed Approach to Address GL 2004-02

The risk associated with GL 2004-02 issues has been quantified as described in Enclosure 3 and is “very small” as defined by Region III in RG 1.174. The proposed FSAR Appendix 6.3A describes the risk-informed approach used to confirm that there is high probability that the ECCS and CSS will perform their design basis functions following a LOCA when considering the impacts and effects described by GL 2004-02. Therefore, no further physical modifications to Callaway are proposed as part of this license amendment request to implement the risk-informed approach.

Attachment 2-1 to this enclosure provides the Licensee Commitment to implement the proposed amendment following approval and to revise affected sections of the plant TS identified in Attachment 2-2 and of the FSAR identified in Attachment 2-5 to this enclosure. Upon approval of the proposed amendment, applicable FSAR safety system and design bases descriptions that take credit for the evaluation described above will be revised. In addition, conforming changes to the TS Bases are provided in Attachment 2-3 to this enclosure for information only, to be implemented following NRC approval of the LAR.

System redundancy, independence, and diversity features are not changed for those safety systems credited in the accident analyses. No new programmatic compensatory activities or reliance on manual operator actions are required to implement this change.

## 3.2 Evaluation

The proposed change meets the current regulations unless it is explicitly related to a requested exemption.

### 3.2.1 Engineering Analysis Overview

The design and licensing basis descriptions of accidents requiring ECCS and CSS operation, including analysis methods, assumptions, and results provided in FSAR Chapters 6 and 15 remain unchanged. This is based on the functionality of the ECCS and CSS during design-basis accidents being confirmed by demonstrating that the calculated risk associated with GL 2004-02 for Callaway is "Very Small" and less than the Region III acceptance guidelines defined by RG 1.174.

In addition, as described in RG 1.174, Section 2.5.2, "Comparisons with Acceptance Guidelines," "if there is an indication that the CDF or LERF could considerably exceed $10^{-4}$ and $10^{-5}$, respectively, in order for the change to be considered, the licensee may need to show why steps should not be taken to reduce CDF or LERF." As shown in the following Table, the current Callaway PRA model of record (Update 9.01) baseline total aggregate risk CDF is 6.74E-05 $yr^{-1}$, and the corresponding LERF is 4.08E-06 $yr^{-1}$.

| **Model** | **Baseline CDF** | **Baseline LERF** |
|---|---|---|
| Internal Events (Excluding Internal Flooding) PRA | 4.46E-06 | 6.23E-08 |
| Internal Flooding PRA | 6.52E-06 | 1.51E-08 |
| Fire PRA | 1.09E-05 | 4.63E-08 |
| Seismic PRA | 4.01E-05 | 4.43E-06 |
| High Winds PRA | 5.40E-06 | 2.50E-07 |
| Other External Events | No significant contribution | No significant contribution |
| Total Aggregate Risk | 6.74E-05 | 4.80E-06 |

From the Section 7, "Baseline Results," in Enclosure 3, Attachment 3-3, it can be seen that the calculated RoverD mean ΔCDF is 5.37E-07, and the corresponding mean ΔLERF is 5.37E-08. Applying the changes in CDF and LERF associated with implementation of RoverD to the baseline CDF and LERF, respectively, the resulting final values are 6.79E-5 $yr^{-1}$ for CDF and 4.85E-6 $yr^{-1}$ for LERF, which are well within the RG 1.174, Section 2.5.2 acceptance guidelines.

The performance evaluations for accidents requiring ECCS operation described in Chapters 6 and 15, based on Callaway 10 CFR 50, Appendix K large-break LOCA analysis, demonstrate that for partial breaks and complete breaks up to and including the DEGB of a reactor coolant pipe, the ECCS will limit the clad temperature to below the limit specified in 10 CFR 50.46, and assure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved.

The LAR is requested for the scope of breaks that can generate fiber debris on the containment sump strainers that exceeds the amount of fiber debris bounded by the plant-specific testing. In Attachment 3-2 and 3-3, Callaway determined that only large breaks on some RCS and RHR pipes were in this scope and has identified 60 weld

break locations (listed in Attachment 3-3). Ameren Missouri is requesting an amendment to the license for this scope of breaks to allow evaluation of the debris effects using a risk-informed methodology because there is no practical deterministic methodology currently available. The amendment is requested to apply to the evaluation methodology and not to the specific set of break locations.

The WCAP-17788 methodology [8] that was used to evaluate the down-stream effects for the deterministic scope of breaks to assure long-term core cooling does not replace the ECCS evaluation methodology described in Callaway FSAR Chapter 15.6. The current Chapter 15.6 LOCA thermal-hydraulic analysis applies only through the LOCA re-flood phase and is not used for the assessment of long-term cooling required by the risk-informed assessment of debris effects.

### 3.2.2 Evaluation of Defense-In-Depth and Safety Margin

Defense-in-Depth Analysis

The proposed change is consistent with the defense-in-depth (DID) philosophy in that the following aspects of the facility design and operation are unaffected:

- Functional requirements and the design configuration of systems
- Existing plant barriers to the release of fission products
- Design provisions for redundancy, diversity, and independence
- Plant's response to transients or other initiating events
- Preventive and mitigative capabilities of plant design features

The proposed amendment does not involve a change in any functional requirements or the configuration of plant SSCs.

Enclosure 3, Attachment 3-4 provides a more detailed description of the defense-in-depth measures that address potential sump blockage and in-core effects, including the means available to operators for detecting and mitigating inadequate recirculation flow and inadequate core cooling flow. The proposed change does not involve a change in any functional requirements, the configuration, or method of performing functions of plant SSCs. The effects from a full spectrum of LOCAs, including a range of partial breaks and DEGBs for all piping sizes up to and including the largest pipe in the reactor coolant system, are analyzed. Appropriate redundancy and consideration of loss of offsite power and worst case single failure are retained. This approach ensures that DID is maintained.

Safety Margin Analysis

Enclosure 3, Attachment 3-4 provides a more detailed discussion on how sufficient safety margins associated with the design are maintained by the proposed change. Approval of the proposed change would add the results of a risk-informed evaluation to the FSAR that concludes that there is high probability that the containment sumps will perform their design basis functions in support of ECCS and CSS recirculation modes following a LOCA when considering the impacts and effects of debris on sump strainers. Core flow blockage due to in-vessel effects is addressed via an analysis using WCAP-17788 methodology. The proposed change does not result in any changes to the safety analyses demonstrating safety margin for the barriers to the release of radioactivity as described in the FSAR, and does not involve a change in the functional requirements, configuration, or method of performing functions of plant SSCs.

### 3.2.3 Description of the PRA

The Callaway PRA is an all-hazards, at-power Level I with large early release frequency (LERF) model. The Callaway PRA is representative of the as-built, as-operated plant and includes the internal events that are within the focus of the GL 2004-02 concerns related to LOCA. The Callaway PRA is reviewed for consistency with the as-built, as-operated plant at a nominal frequency of every two refueling cycles.

The PRA was not changed to address GL 2004-02 concerns. Instead, a detailed engineering analysis was performed in an uncertainty quantification framework that evaluates the required failure modes of ECCS and core cooling (in-vessel effects). Significant detail is included in the engineering analysis, including physical models and mechanisms known to lead to failure, and the analyses account for experimental evidence used to support particular areas of concern.

Callaway's PRA is compliant with RG 1.200, Revision 3 for internal events and is therefore acceptable to support the assessment of the risk of internal events associated with GL 2004-02.

Enclosure 3, Attachment 3-3 to this letter provides a more detailed description of how the Callaway PRA results are used in conjunction with the detailed engineering analysis to estimate ΔLERF and address secondary line break initiating events.

### 3.2.4 Implementation and Monitoring Program

Design modifications addressing GL 2004-02 concerns, including installation of new sump strainers and replacement of some problematic insulation, have been previously implemented using the Callaway design change process.

Callaway has implemented procedures and programs for monitoring, controlling and assessing changes to the plant that have a potential impact on plant performance related to GL 2004-02 concerns. These provide the capability to monitor the performance of the sump strainers and the ability to assess impacts to the inputs and assumptions used in the PRA and the associated engineering analysis that support the

proposed change. Programmatic requirements ensure that the potential for debris loading on the sumps does not materially increase. These include:

- Programs and procedures have been implemented to evaluate and control sources of debris in containment:
  - Technical Requirements implemented by Callaway procedures require visual inspections of all accessible areas of the containment to check for loose debris, and each containment sump to check for debris.
  - The Callaway Engineering Change Control Procedure includes provisions for managing debris sources such as insulation, qualified coatings, addition of aluminum or zinc. The procedure has been augmented as applicable to require changes that involve any work or activity inside the containment be evaluated for the potential to affect the following:
    - Reactor coolant pressure boundary integrity
    - Accident or post-accident equipment inside containment
    - Quantity of metal inside containment
    - Quantity or type of coatings inside containment
    - Thermal insulation changed or added
    - Addition or deletion of cable
  - The 10 CFR 50.59 change process will be entered in accordance with Callaway procedures for all Design Changes. This process ensures that new insulation material that may differ from the initial design is evaluated for GL 2004-02 concerns.
  - Programs to ensure that Service Level 1 protective coatings used inside containment are procured, applied, and maintained in compliance with applicable regulatory requirements. Additional details are discussed in the Callaway coatings program developed in response to Generic Letter 98-04 [9].
  - Procedures have been implemented to govern the use of signs and labels inside containment.
- As a necessary and required support function for ECCS and CSS, the sump strainers are within the Callaway 10 CFR 50.65 Maintenance Rule program:

- As part of the Callaway Corrective Action Program, condition reports written due to adverse conditions identified during the containment inspections or containment sumps and strainers surveillances are reviewed for impact on Maintenance Rule scoped systems, as appropriate.

- The Callaway Maintenance Rule program includes performance monitoring of functions associated with ECCS and CSS, including sump recirculation. The inclusion of the ECCS and CSS into the Maintenance Rule program and the assessment of acceptable system performance provide continued assurance of the availability for performance of the required functions.

- PRA Updates: For the purpose of monitoring future facility changes or other conditions that may impact the PRA results associated with GL 2004-02, appropriate changes to the as-built, as-operated plant are reflected in updates to the Callaway at-power PRA reference model. The Callaway PRA Program is a living program and, as such, is subject to periodic review and updates. These PRA model periodic updates are performed in accordance with Callaway procedures. The effect of changes incorporated into the at-power PRA model of record are assessed to ensure the results of the analysis used to close GL 2004-02 remain within the aggregate baseline acceptance criteria in RG 1.174.

- Licensed Operator Training: Licensed Operators are trained on indications of and actions in response to sump blockage issues related to GL 2004-02, and performance is evaluated during training scenarios designed to simulate plant responses.

  Operator actions required to respond to sump clogging are currently described in ECA-1.3, and operators are trained on implementing defense-in-depth actions (e.g., alternate flow paths) as a part of the Licensed Operator program. Indications of sump blockage are included as part of the Licensed Operator training administered for Emergency Operating Procedure (EOP) performance of switchover activities in addition to general familiarization with the indications of loss of pump suction. Licensed Operator Training includes the monitoring of operating ECCS and CSS pumps during the evolution for transfer to cold-leg recirculation. Operator training also includes actions required on a total loss of sump recirculation capability.

- Quality Assurance (QA): The Callaway Operating Quality Assurance Program (OQAP) is implemented and controlled in accordance with policies, manuals, procedures and the Operating Quality Assurance Manual (OQAM). This program is applicable to safety related structures, systems and components to an extent consistent with their importance to safety. The OQAP complies with the requirements of 10 CFR 50, Appendix B and other program commitments [OQAP Introduction].

The QA Program is implemented with documented instructions, procedures, and drawings which include appropriate quantitative and qualitative acceptance criteria for determining that prescribed activities have been satisfactorily accomplished [OQAM 5.1]. Procedures control the sequence of required inspections, tests, and other operations when important to quality [OQAM 14.4]. To change these controls, the individual procedure must be changed and a similar level of review and approval given to the original procedure is required [OQAM 5.2.1, 5.2.2, 6.3]. Such instructions, procedures, and drawings are reviewed and approved for compliance with requirements appropriate to their safety significance [OQAM 5.1, 5.2 - 5.9].

QA program controls are applied to safety-related SSCs to provide a high degree of confidence that they perform safely and activities are performed as expected [OQAM 2.1, 2.4, 2.5, and 2.6]. The rigorous controls imposed by the QA program provide adequate quality control elements to ensure system component reliability for the required functions.

- Callaway has adopted other programs that help provide early detection and mitigation of leakage in other applications. The proposed change does not involve any changes to ASME Section XI inspection programs or mitigation strategies that have been shown effective in early detection and mitigation of weld and material degradation in Class 1 piping.

### 3.2.5 Technical Evaluation Conclusion

The technical evaluation results demonstrate that the calculated risk associated with GL 2004-02 concerns for Callaway is very small and lie in Region III defined by RG 1.174. Acceptable containment sump design in support of ECCS and CSS during design-basis accidents is confirmed by demonstrating with high confidence that sufficient safety margin and defense-in-depth are maintained. Additional details can be found in Enclosure 3 of this submittal.

## 4. REGULATORY EVALUATION

### 4.1 Applicable Regulatory Requirements/Criteria

The following regulations apply to the proposed amendment. Approval of the proposed amendment is contingent upon approval of the requests for exemptions from these regulations as provided and justified in Enclosure 1.

Regulatory Requirement 10 CFR 50.46(a)(1)

The regulation 10 CFR 50.46(a)(1), states:

> *Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be*

*provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model. This section does not apply to a nuclear power reactor facility for which the certifications required under §50.82(a)(1) have been submitted.*

*(ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models.*

The regulatory requirements of 10 CFR 50.46(a)(1) remain applicable to the model of record that meets the required features of 10 CFR 50, Appendix K. This evaluation model remains the licensing basis for demonstrating that the ECCS calculated cooling performance following postulated LOCAs does not exceed the acceptance criteria.

The proposed changes do not result in any physical changes to the facility or changes to the operation of the plant, and does not change any of the programmatic requirements. Based on demonstrating acceptable LOCA debris mitigation and containment sump and ECCS design for amending the current licensing basis for 10 CFR 50.46(a)(1) as described above, compliance with other regulatory requirements that rely on acceptable design for these systems and components continue to be met in the current licensing basis.

Regulatory Requirement 10 CFR 50 Appendix A

- GDC 35, "Emergency core cooling," states that a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a

rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

- GDC 38, "Containment heat removal," states that a system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

  Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

- GDC 41, "Containment atmosphere cleanup," states that systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

The proposed changes do not affect compliance with these regulations or guidance and will ensure that the lowest functional capabilities or performance levels of equipment required for safe operation are met.

NRC Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," provides the NRC staff's recommendations for using risk information in support of licensee-initiated Licensing Basis changes to a nuclear power plant that require NRC review and approval. This regulatory guide describes an acceptable approach for assessing the nature and impact of proposed Licensing Basis changes by considering engineering issues and applying risk insights.

In implementing risk-informed decision making, Licensing Basis changes are expected to meet a set of key principles. These principles include the following:

1. *The proposed change meets the current regulations unless it is explicitly related to a requested exemption (i.e., a specific exemption under 10 CFR 50.12, "Specific Exemptions')*.

   The exemption request in Enclosure 1 to this letter implements this principle.

2. *The proposed change is consistent with a defense-in-depth philosophy*.

   The proposed change is consistent with the DID philosophy in that the following aspects of the facility design and operation are unaffected:

   - Functional requirements and the design configuration of systems
   - Existing plant barriers to the release of fission products
   - Design provisions for redundancy, diversity, and independence
   - Plant's response to transients or other initiating events
   - Preventive and mitigative capabilities of plant design features

   The Callaway risk-informed approach analyzes a full spectrum of LOCAs, including a range of partial breaks and DEGBs for all piping sizes up to and including the largest pipe in the RCS. By requiring that mitigative capability be maintained in a realistic and risk-informed evaluation of GL 2004-02 for a full spectrum of LOCAs, the approach ensures that defense-in-depth is maintained.

3. *The proposed change maintains sufficient safety margins*.

   The proposed change does not involve a change in any functional requirements or the configuration of plant SSCs. The safety analyses in the FSAR are unchanged. Therefore, sufficient safety margins associated with the design will be maintained by the proposed change.

4. *When proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement*.

   The proposed change is defined as the risk associated with GL 2004-02 concerns. Using engineering analysis and the PRA this risk has been calculated and shown to be less than the threshold for Region III, "Very Small Changes," and is therefore consistent with the Commission's Safety Goal Policy Statement.

5. *The impact of the proposed change should be monitored using performance measurement strategies*.

   Section 3.2.4 of this Enclosure describes the programmatic requirements that ensure the potential for debris loading on the sump does not materially increase. As noted in section 3.2.4, the effect of changes incorporated into the at-power PRA model of record are periodically assessed to ensure the results of the analysis used to close GL 2004-02 remain within the aggregate baseline acceptance criteria in RG 1.174.

NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," describes one acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. The Callaway PRA model used for the risk-informed approach for addressing GL 2004-02 concerns is in compliance with Revision 3 of RG 1.200.

The proposed changes do not affect compliance with these regulatory guides and will ensure that the lowest functional capabilities or performance levels of equipment required for safe operation are met.

### 4.2 Precedent

The NRC plans to use South Texas Plant (STP) Units 1 and 2 as a pilot for other licensees choosing a risk-informed approach for closure of GSI-191 (SECY-12-0093 [10] and ACRS letter to NRC "Safety Evaluation of License Amendment Request by South Texas Project Nuclear Operating Company to Adopt a Risk-informed Resolution of Generic Safety Issue-191" [11]). The STP-piloted risk-informed approach resulted in substantial benefit to both the NRC and industry in support of the development and implementation of risk-informed resolution of GSI-191.

The proposed amendment and accompanying exemption requests provided an approach for other licensees to revise their Licensing Basis in order to close GSI-191 and GL 2004-02.

By Reference 11, the ACRS approved the NRC's safety evaluation of the STPNOC's RoverD analysis results showing that the risks, CDF and large early release frequency (LERF) associated with GSI-191 concerns are less than the threshold for Region III, "Very Small Changes," of RG 1.174, and the NRC accepted the associated license amendment request.

It is Ameren Missouri's intent to follow that pilot program with the following exceptions, which deviate from STPNOC's submittal:

- References to “GSI-191” were replaced with “GL 2004-02” in most places due to the administrative closing of GSI-191 in July 2019.

- Ex-vessel downstream effects are not addressed in the Callaway risk-informed evaluation of GL 2004-02. Instead Callaway performed an analysis based on WCAP-16406 [12].

- In-vessel downstream effects are not addressed in the Callaway risk-informed evaluation of GL 2004-02. Instead, Callaway performed an analysis based on WCAP-17788.

- No extensive thermal-hydraulic analysis of the core’s response to postulated blockage was conducted.

- Completion times for proposed new TS 3.6.8 were based on TSTF-567, Revision 1, which applies a 90-day window to bring the ECCS strainer back to operable condition.

### 4.3 No Significant Hazards Consideration Determination

Callaway has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

**1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

**Response: No.**

The proposed changes involve a methodology change for assessment of debris effects that adds the results of a risk-informed evaluation to the Callaway licensing basis. Included are proposed changes to the Callaway Technical Specifications, for which the Emergency Core Cooling System (ECCS) Technical Specifications would be revised and a new Technical Specification (TS) dedicated to the containment sumps would be established. The new TS would include Conditions and Required Actions for addressing containment conditions involving potential LOCA debris-related effects. The Required Action related to potential LOCA-generated debris effects includes a Completion Time that is longer than the Completion Time currently specified under the ECCS and Containment Spray

System (CSS) Technical Specifications, but is justified on the basis of very low risk. Associated administrative TS changes are proposed as well.

The methodology change (to be reflected in the FSAR) concludes that the ECCS and Containment Spray System (CSS) will have sufficient defense-in-depth and safety margin and that there is high confidence that these systems will perform their design basis functions following a loss-of-coolant accident (LOCA) when considering the impacts and effects of debris accumulation on containment sump strainers in recirculation mode, as well as core flow blockage due to in-vessel effects, following loss-of-coolant accidents. The methodology change also supports the proposed TS changes.

There is no significant increase in the probability of an accident previously evaluated. The proposed changes address mitigation of loss-of-coolant accidents and have no effect on the probability of the occurrence of a LOCA. The proposed methodology and TS changes do not implement any physical changes to the facility or any SSCs and do not implement any changes in plant operation that could lead to a different kind of accident.

The proposed changes do not involve a significant increase in the consequences of an accident previously evaluated. The methodology change confirms that required SSCs supported by the containment sumps will perform their safety functions with a high probability, as required, and does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an accident previously evaluated within the acceptance limits. The safety analysis acceptance criteria in the FSAR continue to be met for the proposed methodology change. The evaluation of the changes determined that containment integrity will be maintained. The dose consequences were considered in the assessment, and quantitative evaluation of the effects on dose using input from the risk-informed approach shows the increase in dose consequences is very small.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of any the accident previously evaluated in the FSAR.

**2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

**Response: No.**

The proposed changes are a methodology change for assessment of debris effects from LOCAs that are already evaluated in the Callaway FSAR, establishment of a new TS dedicated to the containment sumps that addresses potential LOCA-generated debris effects on the ECCS and CSS, and associated administrative changes to the TS. No new or different kind of accident is being evaluated. None of the changes install or remove any plant equipment, or alter the design, physical configuration, or mode of operation of any plant structure, system or component.

The proposed changes do not introduce any new failure mechanisms or malfunctions that can initiate an accident. The proposed changes do not introduce failure modes, accident initiators, or equipment malfunctions that would cause a new or different kind of accident.

Therefore, the proposed changes do not create the possibility for a new or different kind of accident from any previously evaluated.

**3. Does the proposed change involve a significant reduction in a margin of safety?**

**Response: No.**

The proposed changes are a methodology change for assessment of debris effects from LOCAs that are already evaluated in the Callaway FSAR, establishment of a new TS dedicated to the containment sumps that addresses potential LOCA-generated debris effects on the ECCS and CSS, and associated administrative changes to the TS. The effects from a full spectrum of LOCAs, including a range of partial breaks and DEGBs for all piping sizes up to and including the largest pipe in the reactor coolant system, are analyzed. Appropriate redundancy and consideration of loss of offsite power and worst case single failure are retained, such that defense-in-depth is maintained.

Application of the risk-informed methodology showed that the increase in risk from the contribution of debris effects is “very small” as defined by RG 1.174 and that there is adequate defense-in-depth and safety margin. Consequently, Callaway determined that the risk-informed method demonstrates the containment sumps will continue to support the ability of safety-related components to perform their design functions when the effects of debris are considered. The proposed change does not alter the manner in which safety limits are determined or acceptance criteria associated with a safety limit. The proposed change does not implement any changes to plant operation and does not significantly affect SSCs that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. The proposed change does not significantly affect the existing safety margins in the barriers for the release of radioactivity. There are no changes to any of the safety analyses in the FSAR.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Callaway concludes that the proposed changes do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

### 4.4 Conclusions

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations contingent upon approval of the exemption requested in Enclosure 1 to this letter, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 5. ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

## 6. REFERENCES

1. Generic Safety Issue 191, "Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance."
2. Nuclear Regulatory Commission Generic Letter 2004-02: “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,” September 13, 2004 (ML042360586).
3. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3. U.S. Nuclear Regulatory Commission, January 2018 (ML17317A256).
4. Letter from NRC to TSTF, “Final Safety Evaluations of Technical Specifications Task Force Traveler TSTF-567, Revision 1, ‘Add Containment Sump TS to Address GSI-191 Issues’ (EPID: L-2017-PMP-0005,” July 3, 2018 (ML18109A077).
5. NUREG-1829 Vol. 1, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” April 2008 (ML082250436).
6. NRC Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors.” Nuclear Regulatory Commission, June 9, 2003.

7. SECY-10-0113, “Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance,” August 26, 2010.
8. WCAP-17788-NP, “Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090),” Volume 1, Revision 0. Westinghouse, July 2015 (ML15210A669).
9. Generic Letter No. 98-04, “Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment.” Nuclear Regulatory Commission, July 14, 1998.
10. SECY-12-0093, “Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance,” July 9, 2012.
11. ACRS letter to NRC, "Safety Evaluation of License Amendment Request by South Texas Project Nuclear Operating Company to Adopt a Risk-informed Resolution of Generic Safety Issue-191," May 17, 2017 (ML17137A325).
12. WCAP-16406-P, “Evaluation of Downstream Sump Debris Effects in Support of GSI-191 (PA-SEE-0195),” Revision 0. Westinghouse, August 2007.

# Attachment 2-1

# List of Regulatory Commitments

**List of Commitments**

The following table identifies the actions to which the licensee, Ameren Missouri, has committed. Statements in the submittal with the exception of those in the table below are provided for information purposes and are not considered regulatory commitments.

| **Commitment** | **TYPE** | | **Scheduled Completion Date (if applicable)** |
|---|---|---|---|
| | **One-Time** | **Continuing Compliance** | |
| [1.] Ameren Missouri shall complete all work required for final resolution of GL 2004-02 (i.e., all licensing actions such as FSAR, TS and TS Bases changes) per the schedule agreed upon between the industry (NEI) and NRC.<br><br>Commitment number 50262 | X | | 120 days following issuance of NRC Safety Evaluation of Callaway LAR |

# Attachment 2-2

# Technical Specification Page Markups

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SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.2.4 | Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head. | In accordance with the INSERVICE TESTING PROGRAM |
| SR 3.5.2.5 | Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. | In accordance with the Surveillance Frequency Control Program |
| SR 3.5.2.6 | Verify each ECCS pump starts automatically on an actual or simulated actuation signal. | In accordance with the Surveillance Frequency Control Program |
| SR 3.5.2.7 | Verify, for each ECCS throttle valve listed below, each mechanical position stop is in the correct position.<br>Valve Number<br>EMV0095 EMV0107 EMV0089<br>EMV0096 EMV0108 EMV0090<br>EMV0097 EMV0109 EMV0091<br>EMV0098 EMV0110 EMV0092 | In accordance with the Surveillance Frequency Control Program |
| SR 3.5.2.8 | ~~Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.~~ | ~~In accordance with the Surveillance Frequency Control Program~~ |

Not used.

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.3.1 | The following SRs are applicable for all equipment required to be OPERABLE:<br><br>SR 3.5.2.1<br>SR 3.5.2.3<br>SR 3.5.2.4<br>SR 3.5.2.7<br>~~SR 3.5.2.8~~ | In accordance with applicable SRs |

## 3.6 CONTAINMENT SYSTEMS

### 3.6.8 Containment Sumps

LCO 3.6.8 Two containment sumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more containment sumps inoperable due to containment accident generated and transported debris. | A.1 | Initiate actions to reduce containment accident generated and transported debris. | Immediately |
| | AND | | |
| | A.2 | Perform SR 3.4.13.1 | Once per 24 hours |
| | AND | | |
| | A.3 | Restore the containment sump(s) to OPERABLE status. | 90 days |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. One or more containment sumps inoperable for reasons other than Condition A. | B.1 ---------NOTES---------<br>1. Enter applicable Conditions and Required Actions of LCO 3.5.2, "ECCS – Operating," and LCO 3.5.3, "ECCS – Shutdown" for emergency core cooling trains made inoperable by the containment sump(s).<br>2. Enter applicable Conditions and Required Actions of LCO 3.6.6 "Containment Spray and Cooling Systems" for containment spray trains made inoperable by the containment sump(s).<br>----------<br>Restore the containment sump(s) to OPERABLE status. | 72 hours |
| C. Required Action and associated Completion Time not met. | C.1 Be in MODE 3.<br>AND<br>C.2 Be in MODE 5. | 6 hours<br><br>36 hours |

## SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.8.1 | Verify, by visual inspection, the containment sumps do not show structural damage, abnormal corrosion, or debris blockage. | In accordance with the Surveillance Frequency Control Program |

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995, as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

(continued)

# Attachment 2-3

# Technical Specification Bases Page Markups

# (for information only)

TABLE OF CONTENTS (Continued)

CHAPTER TABLE OF CONTENTS (Continued)

and LCO 3.6.8, “Containment Sumps,” and



BASES

BACKGROUND (continued)

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed selected loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

, and the containment sumps, are

,

The active ECCS components, along with the passive accumulators ~~and~~ the RWST covered in LCO 3.5.1, "Accumulators," ~~and~~ LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The ECCS centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.7 (continued)

the other cold legs receive at least the required minimum flow. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The ECCS throttle valves are set to ensure proper flow resistance and pressure drop in the piping to each injection point in the event of a LOCA. Once set, these throttle valves are secured with locking devices and mechanical position stops. These devices help to ensure that the following safety analyses assumptions remain valid: (1) both the maximum and minimum total system resistance; (2) both the maximum and minimum branch injection line resistance; and (3) the maximum and minimum ranges of potential pump performance. These resistances and pump performance ranges are used to calculate the maximum and minimum ECCS flows assumed in the LOCA analyses of Reference 3.

SR 3.5.2.8

Not used.

~~Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.

2. 10 CFR 50.46.

3. FSAR, Sections 6.3 and 15.6.

4. FSAR, Chapter 15, "Accident Analysis."

5. NRC Memorandum to V. Stello, Jr., from R. L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

6. IE Information Notice No. 87-01.

7. RFR-14801A.

8. ULNRC-2535 dated 12-18-91 (for SI and RHR pumps) and ULNRC-04583 dated 12-13-01 (for CCPs).

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

and the containment sumps

BASES

BACKGROUND

The function of the ECCS in MODE 4 is to provide core cooling to ensure that the reactor core is protected after a small break loss of coolant accident (SBLOCA). In MODE 4, the required ECCS train consists of two sperate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following a SBLOCA.

During low temperature conditions is in RCS, limitations are placed on the maximum number of ECCS pumps that are capable of injecting into the RCS. Refer to the Bases for LCO 3.4.12, “Cold Overpressure Mitigation System (COMS),’ for the basis of these requirements.

The ECCS components in MODE 4 provide the cooling water necessary to meet GDC 35 (Reference 1).

APPLICABLE SAFETY ANALYSES

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a SBLOCA, the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a SBLOCA.

The LCO helps to ensure that the following acceptance criteria for the ECCS, established be 10 CFR 50.46 (Reference 2), will be met following a SBLOCA:

a. Maximum fuel element cladding temperature is $\leq$ 2200°F;

b. Maximum cladding oxidation is < 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is $\leq$ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

(continued)

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.8 Containment Sumps

BASES

BACKGROUND

The containment sumps (described in the Final Safety Analysis Report as "containment recirculation sumps") provide a borated water source to support recirculation of coolant from the containment sumps for emergency core cooling and containment spray cooling during the long-term phase of an accident.

The containment sumps supply each train of the Emergency Core Cooling System (ECCS) and the Containment Spray System (CSS) during any accident that requires recirculation of coolant from the containment sumps. The recirculation mode is initiated when the residual heat removal (RHR) pump suction is automatically transferred to the containment sumps on Lo-Lo-1 Refueling Water Storage Tank (RWST) level, which ensures the containment sumps have enough water to supply the net positive suction head to the ECCS and CSS pumps. The design uses two containment sumps to supply both trains of the ECCS and CSS independently, i.e., each sump supplies a complete train of ECCS and CSS.

The containment recirculation sumps contain strainers to limit the quantity of the debris materials from entering the sump suction piping. Debris accumulation on the strainers can lead to undesirable hydraulic effects including air ingestion through vortexing or deaeration, and reduced net positive suction head (NPSH) at pump suction piping.

While the majority of debris accumulates on the strainers, some fraction penetrates the strainers and is transported to downstream components in the ECCS, CSS, and the Reactor Coolant System (RCS). Debris that penetrates the strainers can result in wear to the downstream components, blockages, or reduced heat transfer across the fuel cladding. Excessive debris in the containment sump water source could result in insufficient recirculation of coolant during the accident, or insufficient heat removal from the core during the accident.

BASES

APPLICABLE SAFETY ANALYSES

During all accidents that require recirculation, the containment sumps provide a source of borated water to the ECCS and CSS pumps. As such, the containment sumps support emergency core cooling and containment cooling during an accident. They also provide a source of negative reactivity. The design basis transients and applicable safety analyses concerning the containment sumps are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating," B 3.5.3, "ECCS - Shutdown," and B 3.6.6, "Containment Spray and Cooling Systems."

FSAR Section 15 describes evaluations that confirm long-term core cooling is assured following any accident that requires recirculation from the containment sumps.

The containment sumps satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The containment sumps are required to ensure a source of borated water to support ECCS and CSS OPERABILITY. A containment sump consists of the in-containment ECCS and CSS recirculation flow paths, (including debris interceptors, trash racks, and curbs), the containment sump strainer, and the inlet to the ECCS and CSS piping. An OPERABLE containment sump has no structural damage or abnormal corrosion that could prevent recirculation of coolant and will not be restricted by containment accident generated and transported debris.

Callaway's containment sump design includes more than one containment sump. Two sumps are considered part of a single support system because containment accident generated and transported debris issues that could render one sump inoperable could render all of the sumps inoperable. When a supported system LCO is not met solely due to a support system LCO not being met, only the support system LCO ACTIONS are required to be entered.

Containment accident generated and transported debris consists of the following:

a. Accident-generated debris sources – Insulation, coatings, and other materials assumed to be damaged by the postulated high-energy line break (HELB) and transported to a containment sump. This includes materials within the HELB zone of influence and other materials (e.g., unqualified coatings) that fail due to the post-accident containment environment following the accident;

(continued)

BASES

LCO (continued)

b. Latent debris sources – Pre-existing dirt, dust, paint chips, fines or shards of insulation, and other materials inside containment that do not have to be damaged by the HELB to be transported to a containment sump; and

c. Chemical product debris sources – Aluminum and nonmetallic materials such as paints, thermal insulation, and concrete that are susceptible to chemical reactions within the post-accident containment environment leading to corrosion products that are generated within a containment sump pool or are generated within containment and transported to the containment sumps.

Containment debris limits are defined in FSAR Chapter 6, Table 6.3-13.

APPLICABILITY

In MODES 1 and 2, 3, and 4, containment sump OPERABILITY requirements are dictated by the ECCS and CSS OPERABILITY requirements. Since both the ECCS and the CSS must be OPERABLE in MODES 1, 2, 3, and 4, the containment sumps must also be OPERABLE to support their operation.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the containment sumps are not required to be OPERABLE in MODES 5 or 6.

ACTIONS

A.1, A.2, and A.3

Condition A is applicable when there is a condition which results in containment accident generated and transported debris exceeding the analyzed limits. Containment debris limits are defined in FSAR Chapter 6, Table 6.3-13.

Immediate action must be initiated to mitigate the condition. Examples of mitigating actions are:

- Removing the debris source from containment or preventing the debris from being transported to the containment sump;
- Evaluating the debris source against the assumptions in the analysis;
- Deferring maintenance that would affect availability of the affected systems and other LOCA-mitigating equipment;
- Briefing operators on LOCA debris management actions; or

(continued)

BASES

ACTIONS (continued)

- Applying an alternative method to establish new limits.

While in this condition, the RCS water inventory balance, SR 3.4.13.1, must be performed at an increased Frequency of once per 24 hours. An unexpected increase in RCS leakage could be indicative of an increased potential for an RCS pipe break, which could result in debris being generated and transported to the containment sumps. The more frequent monitoring allows operators to minimize the potential for an RCS pipe break proactively while a containment sump is inoperable.

The inoperable containment sumps must be restored to OPERABLE status in 90 days. A 90-day Completion Time is reasonable for emergent conditions that involve debris in excess of the analyzed limits that could be generated and transported to the containment emergency sump under accident conditions. The likelihood of an initiating event in the 90-day Completion Time is very small, and there is safety margin in the associated analyses. The mitigating actions of Required Action A.1 provide additional assurance that the effects of debris in excess of the analyzed limits will be mitigated during the Completion Time.

B.1

When a containment sump(s) is inoperable for reasons other than Condition A, such as blockage, structural damage, or abnormal corrosion that could prevent recirculation of coolant, it must be restored to OPERABLE status within 72 hours. The 72-hour Completion Time takes into account the reasonable time for repairs and low probability of an accident that requires the containment sumps occurring during this period.

Required Action B.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.5.2, "ECCS - Operating," and LCO 3.5.3, "ECCS - Shutdown," should be entered if an inoperable containment sump(s) results in an inoperable ECCS train. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.6.6, "Containment Spray and Cooling Systems," should be entered if an inoperable containment sump(s) results in an inoperable CSS train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the containment sump(s) cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Periodic inspections are performed to verify the containment sumps do not show current or potential debris blockage, structural damage, or abnormal corrosion to ensure the operability and structural integrity of the containment sump, strainers, and associated structures.

A containment sump consists of the in-containment ECCS and CSS recirculation flow paths, (including debris interceptors, trash racks, and curbs), the containment sump strainer, and the inlets to the ECCS and CSS piping.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Chapter 6 and Chapter 15.

# Attachment 2-4

# Re-typed Technical Specification Pages

TABLE OF CONTENTS

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.2.4 | Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head. | In accordance with the INSERVICE TESTING PROGRAM |
| SR 3.5.2.5 | Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. | In accordance with the Surveillance Frequency Control Program |
| SR 3.5.2.6 | Verify each ECCS pump starts automatically on an actual or simulated actuation signal. | In accordance with the Surveillance Frequency Control Program |
| SR 3.5.2.7 | Verify, for each ECCS throttle valve listed below, each mechanical position stop is in the correct position.<br>Valve Number<br>EMV0095 EMV0107 EMV0089<br>EMV0096 EMV0108 EMV0090<br>EMV0097 EMV0109 EMV0091<br>EMV0098 EMV0110 EMV0092 | In accordance with the Surveillance Frequency Control Program |
| SR 3.5.2.8 | Not used. | |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.3.1 | The following SRs are applicable for all equipment required to be OPERABLE:<br><br>SR 3.5.2.1 SR 3.5.2.7<br>SR 3.5.2.3<br>SR 3.5.2.4 | In accordance with applicable SRs |

## 3.6 CONTAINMENT SYSTEMS

### 3.6.8 Containment Sumps

LCO 3.6.8 Two containment sumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more containment sumps inoperable due to containment accident generated and transported debris. | A.1 | Initiate actions to reduce containment accident generated and transported debris. | Immediately |
| | AND | | |
| | A.2 | Perform SR 3.4.13.1 | Once per 24 hours |
| | AND | | |
| | A.3 | Restore the containment sump(s) to OPERABLE status. | 90 days |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. One or more containment sumps inoperable for reasons other than Condition A. | B.1 ---------NOTES---------<br>1. Enter applicable Conditions and Required Actions of LCO 3.5.2, "ECCS – Operating," and LCO 3.5.3, "ECCS – Shutdown" for emergency core cooling trains made inoperable by the containment sump(s).<br><br>2. Enter applicable Conditions and Required Actions of LCO 3.6.6 "Containment Spray and Cooling Systems" for containment spray trains made inoperable by the containment sump(s).<br>----------<br>Restore the containment sump(s) to OPERABLE status. | 72 hours |
| C. Required Action and associated Completion Time not met. | C.1 Be in MODE 3.<br><br>AND<br><br>C.2 Be in MODE 5. | 6 hours<br><br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.8.1 | Verify, by visual inspection, the containment sumps do not show structural damage, abnormal corrosion, or debris blockage. | In accordance with the Surveillance Frequency Control Program |

## 5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995, as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

# Attachment 2-5

# Final Safety Analysis Report Page Markups

(for information only)

## Callaway FSAR Page Markups

Upon approval of the licensing basis changes, Ameren Missouri will add a new Appendix 6.3A, “Risk-Informed Approach to Potential Impact of Debris Blockage on Emergency Recirculation during Design-Basis Accidents,” to the FSAR. Appendix 6.3A will describe the evaluations performed using a risk-informed approach to address GSI-191 concerns including the effects on long-term cooling due to debris accumulation on containment sump strainers for ECCS and CSS in recirculation mode, as well as core flow blockage due to in-vessel effects, following LOCAs. Note that FSAR Appendix 6.3A.2 describes change control requirements, monitoring requirements and reporting requirements which require prior NRC approval for changes, and NRC approval of the requirements described in FSAR Appendix 6.3A.2 is requested as part of this application.

FSAR Table 1.6-2 (WCAPs Incorporated by Reference and Sections 3.1 (GDC, 6.2 (Containment Systems, 6.3 (ECCS, and 15.6 (LOCA will be updated to reference Appendix 6.3A as needed.

TABLE 1.6-2 (Sheet 22)

(1) A legend to the review status code letters follows:

A - NRC review complete; NRC acceptance letter issued.

AE - NRC accepted as part of the Westinghouse emergency core cooling system (ECCS) evaluation model only; does not constitute acceptance for any purpose other than for ECCS analyses.

B - Submitted to NRC as background information; not undergoing formal NRC review.

O - On file with NRC; older generation report with current validity; not actively under formal NRC review.

U - Actively under formal NRC review.

(2) Portions of WCAP 7769 are superceded by the Callaway overpressure protection report included in Westinghouse letter SCP 94-143 dated 8/30/94. This report was reviewed by the NRC during the review of OL-1186, Amendment 128, MSSV tolerance change.

(3) Westinghouse letter SCP-10-31, "Transmittal of Mode 4 Small Break LOCA (SBLOCA) RHR Flow Evaluation for Callaway (SCP), Phase 3, Rev. 1," dated May 11, 2010, provides a Callaway-specific evaulation of WCAP-12476.

(P) - Proprietary

T - NRC Technical Evaluation Report issued in lieu of approval

| Westinghouse Topical Report No. | Title | Revision Number | FSAR Section Reference | Submitted to the NRC | Review(1) Status |
|---|---|---|---|---|---|
| WCAP-16406- P-A (P) | Evaluation of Downstream Sump Debris Effects in Support of GSI-191 (PA-SEE- 0195) | Rev. 0 | 6.3A 15.6.5.5 | 8/07 | A |
| WCAP-17788-P (P) | Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE- 1090) | Rev. 0 | 6.3A 15.6.5.5 | 7/15 | T |

The residual heat removal system, in conjunction with the steam and power conversion system, is designed to transfer the fission product decay heat and other residual heat from the reactor core at a rate which keeps the fuel within acceptable limits. The residual heat removal system functions when temperature and pressure are below approximately 350°F and 400 psig, respectively.

Redundancy of the residual heat removal system is provided by two residual heat removal pumps (located in separate flood-proof compartments, with means available for draining and monitoring leakage), two heat exchangers, and associated piping, cabling, and electric power sources. For a more detailed description of residual heat removal system redundancy, refer to Section 5.4.7. The residual heat removal system is able to operate on either the onsite or offsite electrical power system.

Redundancy of heat removal at temperatures above approximately 350°F is provided by the four steam generators, four atmospheric relief valves, and the auxiliary feedwater system.

Details of the system design are provided in Section 5.4.7.

CRITERION 35 - EMERGENCY CORE COOLING

"A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

"Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

DISCUSSION

An emergency core cooling system has the capability to mitigate the effects of any LOCA within the design bases. Cooling water is provided in an emergency to transfer heat from the core at a rate sufficient to maintain the core in a coolable geometry and to assure that clad metal-water reaction is limited to less than 1 percent. Design provisions assure performance of the required safety functions even with a single failure.

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An exemption to GDC 35 has been approved for Callaway to allow application of a risk-informed analysis instead of the deterministic methods required by GDC 35, with respect to the scope of postulated breaks that can generate and transport debris not bounded by plant-specific deterministic testing. Details of the conditions for the exemption are included in Appendix 6.3A. Those breaks that are not bounded by strainer testing have been evaluated for their risk significance, as described in Appendix 6.3A.

(Refer to the discussion of containment recirculation sumps in Section 6.2).

Regarding the containment spray function, an exemption to GDC 38 has been approved for Callaway to allow application of a risk-informed analysis instead of the deterministic methods required by GDC 38, with respect to the scope of postulated breaks that can generate and transport debris not bounded by plant-specific deterministic testing. Details of the conditions for the exemption are included in Appendix 6.3A. Those breaks that are not bounded by strainer testing have been evaluated for their risk significance, as described in Appendix 6.3A.

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DISCUSSION

The containment spray and containment fan cooler systems, in conjunction with the residual heat removal system, are capable of removing sufficient energy and subsequent decay energy from the containment following the hypothesized LOCA to maintain the containment pressure below the containment design pressure. During the post-accident injection phase, water for the containment spray system and residual heat removal system is drawn from the refueling water storage tank. During the later recirculation phase, spray water and reflood water are pumped from the containment sump.

Each of these systems consists of two independent subsystems supplied from separate IE power busses. No single failure, including loss of onsite or offsite electrical power, can cause loss of more than half of the installed 200 percent cooling capacity. The containment spray system and containment fan coolers are discussed in Chapter 6.0. Electrical facilities are described in Chapter 8.0. A containment pressure and temperature analysis following a LOCA is given in Chapter 6.0 with additional results found in Chapter 15.0.

CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

"The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system."

DISCUSSION

The essential equipment of the containment spray system is outside the containment, except for risers, distribution header piping, spray nozzles, and the containment sump. The containment sump, spray piping, and nozzles can be inspected during shutdown. Portions of the containment spray suction piping and the RHR suction piping from the containment recirculation sumps are embedded in concrete and are not accessible for inspection. A portion of the piping from the refueling water storage tank is buried in the ground and not accessible for inspection. Associated equipment outside the containment can be visually inspected.

The containment air coolers and associated cooling water system piping inside the containment can be inspected during shutdowns.

These periodic inspections assure that the capability of these heat removal systems as specified in the Callaway Technical Specifications is met.

offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure."

DISCUSSION

The containment spray system serves to remove radioiodine and other airborne particulate fission products from the containment atmosphere following a LOCA. The system consists of two independent systems, each supplied from separate electrical power busses, as described in Chapter 8.0. Either subsystem alone can provide the fission product removal capacity for which credit is taken in Chapter 15.0, in compliance with Regulatory Guide 1.4.

The generation of hydrogen in the containment under post-accident conditions has been evaluated, using the assumptions of Regulatory Guide 1.7 (see Chapter 6.0). A post-accident hydrogen recombiner system is provided with redundancy of vital components so that a single failure does not prevent timely operation of the system. This system is described in Section 6.2.5. A hydrogen purge system is provided as a backup. No single failure causes both subsystems to fail to operate.

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

"The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems."

DISCUSSION

Regarding the containment spray function, an exemption to GDC 41 has been approved for Callaway to allow application of a risk-informed analysis instead of the deterministic methods required by GDC 41, with respect to the scope of postulated breaks that can generate and transport debris not bounded by plant-specific deterministic testing. Details of the conditions for the exemption are included in Appendix 6.3A. Those breaks that are not bounded by strainer testing have been evaluated for their risk significance, as described in Appendix 6.3A.

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of the containment spray system and the hydrogen control system located inside the containment can be inspected during refueling shutdowns. See Chapter 6.0 for details on the containment spray system and details of the hydrogen control system.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

"The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the

TABLE OF CONTENTS (Continued)

LIST OF TABLES (Continued)

k. Accident Chronology

The chronology of events occurring after a DEPSG break with minimum safeguards is given in Table 6.2.1-6. The chronology of events after a DEPSG break with maximum safeguards is given in Table 6.2.1-7. The chronology of events occurring after a DEHLG break is given in Table 6.2.1-43.

l. Mass and Energy Balances

A mass and energy balance for the reactor coolant system, steam generators, and the safety injection system is provided in Section 6.2.1.3.2 and shows the distribution of energy prior to the accident, at the end of the blowdown phase, at the end of the core reflood phase, and at the end of the post-reflood phase.

A mass and energy balance for the
the DEPSG break with minimum saf
maximum safeguards are provided i
6.2.1-41, and 6.2.1-42. These table
the following times:

1. Prior to the accident

2. End of blowdown

3. End of reflood

4. End of SG energy release

An exemption to GDC 35 has been approved for Callaway to allow application of a risk-informed analysis instead of the deterministic methods required by GDC 35, with respect to the scope of postulated breaks that can generate and transport debris not bounded by plant-specific deterministic testing. Details of the conditions for the exemption are included in Appendix 6.3A. Those breaks that are not bounded by strainer testing have been evaluated for their risk significance, as described in Appendix 6.3A.

m. Long-Term Cooling Following a LOCA

The long-term system behavior during various LOCAs has been evaluated to verify the ability of the ECCS and the containment heat removal systems to keep the reactor vessel flooded and maintain the containment below design conditions for all times following a LOCA. This evaluation is based on the conservative predictions of the performance of these engineered safety features consistent with the single failures assumed for each accident analyzed. The heat generation rate from shutdown fissions, heavy isotope decay, and fission product decay is provided in Figure 6.2.1-16.

The containment pressure and temperature transients for the DEPSG break with minimum safeguards up to $10^6$ seconds are shown in Figures 6.2.1-1 and 6.2.1-4, respectively. These figures demonstrate the containment systems' capability of rapidly reducing the containment

CONTAINMENT RECIRCULATION SUMPS - The two containment recirculation sumps are collecting reservoirs from which the containment spray pumps and the residual heat removal pumps separately take suction after the contents of the refueling water storage tank have been expended. The sumps are located as far as feasible from the reactor coolant system piping and components which could become sources of debris. Thermal insulation used inside containment will be a source of debris. The majority of insulation is NUKON which is discussed in Reference 2, although a significant amount of NUKON was replaced with metallic reflective insulation on the replacement steam generators. Limited quantities of other types of insulation are used in widely dispersed locations. ~~A design basis accident will not degrade a sufficient quantity of this insulation to adversely affect the performance of the sump.~~ Containment emergency recirculation sump strainers are installed within each sump and prevent floating debris and high-density particles from entering. The strainer perforated plate has nominal 0.045 inch openings. The strainer support structure is designed to keep debris from bypassing the strainer. The strainer arrangement is shown in Figure 6.2.2-3.

Sources of debris, as indicated above, are physically remote from the recirculation sumps. Debris generated as a result of a LOCA will either be retained in an area such as the reactor cavity or refueling pool or must follow a tortuous path to reach the recirculation sump strainers. ~~Therefore, no appreciable debris will reach the recirculation sump strainers to cause any significant blockage.~~ In addition, as demonstrated in Figure 6.2.2-3, the recirculation sumps are covered with concrete pads supporting the accumulator tanks; thus, debris cannot fall directly upon the strainer structure. However, the strainers have been sized per Regulatory Guide 1.82, as discussed in Section 6.2.2.1.3, Safety Evaluation Twelve. To limit any possible vortexing, vortex breakers are placed in the suction lines from containment sumps to the containment spray pumps. The suction pipe from the sump is horizontal to limit any possible vortexing and has sufficient submergence to ensure continuous intake flow. The suction lines from the containment sumps to the containment spray pumps are sloped to assure switchover capability. These lines, up to and including the isolation valve, are encased in guard piping.

DEBRIS BARRIERS AND BASKETS - The debris barriers and baskets are designed to reduce the quantity of debris at the containment sumps following a high energy line break (HELB) inside the bioshield. The barriers will diminish the amount of debris that can take the sho
that debris will be
HELB flood plain
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As described in the Callaway response to GL 2004-02 "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," a risk-informed methodology has been applied to evaluate the risk associated with effects of transported debris. This risk-informed analysis shows that the increase in risk associated with debris that would exceed the design limits of the sump structures is "very small," in accordance with the acceptance criteria of Regulatory Guide 1.174. The risk-informed methodology and results are discussed in Appendix 6.3A. See Table 6.3-13 for containment recirculation sump debris source limits.

Regarding the potential for blockage of the containment recirculation sump strainers due to LOCA-generated debris, a risk-informed evaluation has shown that the risk contribution to ΔCDF and ΔLERF from debris effects on the strainers is very small in accordance with RG 1.174. The risk-informed methodology and results are described in Appendix 6.3A.

heat removal by heat exchangers, as described in Section 5.4.7. Credit is taken for heat removal from heat exchangers during the recirculation phase based on a tube side inlet temperature equal to the recirculation sump temperature, which is given in Section 6.2.1 as a function of time after the accident.

Each spray header train provides a minimum of 90-percent area coverage at the o
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A risk-informed evaluation has shown that the risk contribution to ΔCDF and ΔLERF from debris effects on the containment recirculation sump strainers is very small in accordance with Regulatory Guide 1.174. The risk-informed methodology and results are described in Appendix 6.3A.

containment spray envelope reduction factor to determine the design basis coverage. The minimum of 90-percent area coverage at the operating deck is used as a layout gu n the containment spray headers to assure 10 operating floor of the containment. Physical ob crane, are not considered to impede the sp ce created by the hydrogen mixing fans, containment air coolers, the spray within the containment, and the blowdown resulting from the postulated rupture. Thus, the header layout coupled with the extreme turbulence assures the validity of a one-region model above the operating deck for accident dose calculations (see Chapter 15.0).

The formation of chemical precipitates due to metal and debris interactions with containment spray water following a LOCA is described in Appendix 6.3A.

recirculation

Discussion of the volume of containment covered by the sprays is provided in Section 6.5.2.

SAFETY EVALUATION NINE - That part of the CSS located inside the containment is designed to remain operable in the containment accident environment described in Section 3.11(B). The material compatibility of the containment spray system in contact with the post-accident recirculation fluids is discussed in Section 6.1. That part of the CSS located in the auxiliary building is designed to remain operable in the auxiliary building accident environment described in Section 3.11(B).

SAFETY EVALUATION TEN - The borated spray solution is stable under the anticipated LOCA thermal and radiolytic conditions. The borated solution is chemically compatible with components with which it may come into contact. The use of materials which react to release hydrogen (principally zinc and aluminum) has been minimized in equipment located inside the containment. An analysis of hydrogen generation following a LOCA is given in Section 6.2.5.

SAFETY EVALUATION ELEVEN - System piping size and layout will provide adequate NPSH to the containment spray pump during all anticipated operating conditions, in accordance with Regulatory Guide 1.1. In calculating available NPSH, the conservative assumption has been made that the water in the containment sump after a design basis LOCA is a saturated liquid, and no credit has been taken for anticipated subcooling. That is, although NPSH = elevation head + (containment pressure - liquid vapor pressure) - suction line losses, the (containment pressure - liquid vapor pressure) term has been assumed to be zero. Calculated NPSH exceeds required NPSH by at least 10

recirculation

percent. The recirculation piping penetrating the containment sumps is nearly horizontal to minimize vortexing. In addition, a vortex breaker is provided in the inlet of the piping from the sump.

In calculating the water level within the reactor building which contributes to the NPSH available to the containment spray pumps at the beginning of its recirculation phase, consideration has been given to the potential mechanisms of water loss within the reactor building. These water loss mechanisms include water present in the vapor phase, water loss to compartments below El. 2,000, water loss above El. 2,000, and water loss due to wetted surfaces. Tables 6.2.2-6 and 6.2.2-6a identify each water source which releases water to the reactor building and its associated mass and each potential water loss mechanism and the volume of water not assumed to contribute to the water level within the containment for a large LOCA and a MSLB, respectively. The static head available to contribute to the NPSH of the pump, suction line losses, and the minimum NPSH available are also given in Table 6.2.2-7. The CSS pump NPSH versus flow is shown in Figure 6.2.2-5. The reduction in water level due to potential water loss mechanisms is considered in the calculated NPSH available.

SAFETY EVALUATION TWELVE - Recirculation sump strainer construction provides straining down to 0.045 inch to prevent entrained particles in excess of that size from entering the containment recirculation sump and containment spray system suction piping.

Since the containment spray pumps are designed to operate with entrained particles up to 1/4 inch in diameter and the minimum constriction size in the spray nozzles is 7/16 inch, the strainers are adequate to assure proper system operability.

Each strainer is designed to ensure sufficient NPSH to the containment spray and ECCS pumps to maintain recirculation capability during the recirculation phase of an event. The strainer arrangement is shown in Figure 6.2.2-3.

The strainer arrangement does not allow flow into the sump below 6 inches above the concrete floor level surrounding the sump. This arrangement leaves ample depth for buildup of high-density debris without affecting sump performance. Additionally, the velocity of recirculated fluids approaching the strainer will be less than 0.08 fps for all modes of operation following a LOCA or MSLB, and thus a low velocity settling region for high-density particles is provided. Table 6.2.2-9 provides the approach flow velocity for a large LOCA and an MSLB.

Any debris which eludes the strainers and settling region passes into the sump through the 0.045 inch perforated plate and will be drawn into the suction piping for the containment spray and residual heat removal systems. ~~Such debris is small enough to pass through any restriction in either system or the reactor vessel channels, and will eventually be pumped back into the containment.~~

Engineering analysis shows that such debris will not affect downstream components internal or external to the reactor vessel. A risk-informed evaluation has shown that the risk contribution to ΔCDF and ΔLERF from debris effects on the strainers is very small in accordance with RG 1.174. The risk-informed methodology and results are described in Appendix 6.3A.

TABLE 6.2.2-1 COMPARISON OF THE RECIRCULATION SUMP DESIGN WITH EACH OF THE POSITIONS OF REGULATORY GUIDE 1.82

| | Regulatory Guide 1.82 Position | Recirculation Sump Design |
|---|---|---|
| 1. | A minimum of two sumps should be provided, each with sufficient capacity to serve one of the redundant halves of the ECCS and CS systems. | Two sumps are provided, and each has sufficient capacity to serve one of the redundant halves of the ECCS and CS systems. |
| 2. | The redundant sumps should be physically separated from each other and from high energy piping systems by structural barriers, to the extent practical, to preclude damage to the sump intake filters by whipping pipes or high-velocity jets of water or steam. | The redundant sumps are physically separated from each other and from high energy piping. |
| 3. | The sumps should be located on the lowest floor elevation in the containment exclusive of the reactor vessel cavity. As a minimum, the sump intake should be protected by two screens: (1) an outer trash rack and (2) a fine inner screen. The sump screens should not be depressed below the floor elevation. | The sumps are located ~~in El. +2,000~~ at the 2000' elevation, which is the lowest floor elevation in the reactor building, exclusive of the reactor cavity. The containment recirculation strainers are fabricated from stainless steel perforated plate with stainless reinforcement. The perforated plate is more structurally rigid than screens and precludes the use of trash racks. The strainers are installed in the recirculation sump pit and extend approximately one foot above the 2000' elevation of the Reactor Building. The intent is met. |
| 4. | The floor level in the vicinity of the coolant sump location should slope gradually down away from the sump. | The floor is level in the vicinity of the sump.. However, a 6-inch concrete curb is provided which prevents high density particles from entering the sump. |
| 5. | All drains from the upper regions of the reactor building should terminate in such a manner that direct streams of water, which may contain entrained debris, will not impinge on the filter assemblies. | All drains in the upper regions of the reactor building are terminated in such a manner that direct streams of water which may contain entrained debris will not impinge on the filter assemblies. |

In addition, NRC Generic Letter (GL) 2004-02 "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accident at Pressurized-Water Reactors" required licensees to evaluate the ECCS and CSS recirculation functions based on the potential susceptibility of sump screens to debris blockage during design basis accidents. Refer to Section 6.2.2.1.2.

TABLE 6.2.2-1 (Sh

| | Regulatory Guide 1.82 Position | |
|---|---|---|
| 6. | A vertically mounted outer trash rack should be provided to prevent large debris from reaching the fine inner screen. The strength of the trash rack should be considered in protecting the inner screen from missiles and large debris. | The c fabricated from stainless steel perforated plate, including structural reinforcement, and are sufficiently rigid to preclude the use of a trash rack. The structural evaluation for the strainers concludes that the strainers meet the acceptance criteria for all applicable loadings during the recirculation phase of an event. The sumps and strainers are outside the secondary shield wall which provides protection from missiles and large debris. The intent is met. |
| 7. | A vertically mounted fine inner screen should be provided. The design coolant velocity at the inner screen should be approximately 6 cm/sec (0.2 ft/sec). The available surface area used in determining the design coolant velocity should be based on one-half of the free surface area of the fine inner screen to conservatively account for partial blockage. Only the vertical screens should be considered in determining available surface area. | The containment sump strainers are composed of stainless steel perforated plate with 0.045-inch diameter holes. The approach velocity of the recirculation coolant flow at the sump strainer face is less than 0.2 ft/sec. The intent is met. |
| 8. | A solid top deck is preferable, and the top deck should be designed to be fully submerged after a LOCA and completion of the safety injection. | A concrete slab over the containment sump strainers is provided. The containment recirculation sump strainers will be fully submerged following a large break LOCA or a small break LOCA. |
| 9. | The trash rack and screens should be designed to withstand the vibratory motion of seismic events without loss of structural integrity. | The containment recirculation sump strainers are designed as seismic Category I and have been evaluated acceptably for all applicable loadings. |

TABLE 6.2.2-7 INPUT AND RESULTS OF NPSH ANALYSIS

22.8

16.6

6

26.2

21.8

Containment Spray Pumps

| | |
|---|---|
| Static head available (LOCA) | 31 ft - 3-1/4 in. |
| Pump elevation (discharge centerline) | 1971 ft - 0-3/4 in. |
| Suction line losses @ 3,950 gpm | 9.2 ft |
| Available NPSH @ 3,950 gpm | ~~22.0~~ ft |
| Required NPSH @ 3,950 gpm (from Figure 6.2.2-5) | ~~16.5~~ ft |

Residual Heat Removal Pumps

| | |
|---|---|
| Pump elevation (discharge centerline) | 1971 ft - 9-1/2 in. |
| Static head available (LOCA)(1) | 30 ft - ~~0-1/2~~ in. |
| Suction line losses @ 4,800 gpm | 4.3 ft |
| Available NPSH @ 4,800 gpm | ~~25.7~~ ft |
| Required NPSH @ 4,800 gpm (from Figure 6.3-3) | ~~21.7~~ ft |

(1) Large LOCA conditions are provided for the RHR pumps since the flow rates, line losses, and NPSH required are greater than those associated with an MSLB wherein the RCS pressure remains above the RHR shutoff head at switchover to recirculation.

The values are taken from MP 15-0017.

| | |
|---|---|
| Total Sump Screen Head Loss (max.) | 4.2 ft-$H_2O$ |
| NPSH Margin | 2.0 ft |

that air pressure on the diaphragm overcomes the spring force, causing the linkage to move the butterfly to the closed position. Upon loss of air pressure, the spring returns the butterfly to the open pos
normal operation to maximiz
of the ECCS operation. The
control cooldown flowrate.

In addition, head losses caused by the containment recirculation sump strainers' structure and debris load associated with the limiting line break are included in Tables 6.2.2-7 and 6.3-1. These head losses were used to calculate the NPSH margin and show adequate NPSH for RHR and CS pump operation. Maximum analyzed debris loads associated with this limiting line break are given in Table 6.3-13 and discussed in Appendix 6.3A.

Each RHR heat exchanger b
normally closed and is desig
cooldown to avoid thermal s

Net Positive Suction Head

Available and required net positive suction head (NPSH) for ECCS pumps are shown in Table 6.3-1. Table 6.2.2-7 provides the assumptions and results of the NPSH analyses for the containment spray and RHR pumps. The safety intent of Regulatory Guide 1.1 is met by the design of the ECCS so that adequate NPSH is provided to system pumps. In addition to considering the static head and suction line pressure drop, the calculation of available NPSH in the recirculation mode assumes that the vapor pressure of the liquid in the sump is equal to the containment ambient pressure. This ensures that the actual available NPSH is always greater than the calculated NPSH. To ensure that the required NPSH is available during the recirculation phase of ECCS operation, restriction orifices are provided in the four discharge lines into the RCS cold legs and in the two discharge lines into the RCS hot legs.

~~Accumulator Motor-Operated Valve~~

As part of the plant shutdown administrative procedures, the operator is required to close these valves. This prevents a loss of accumulator water inventory to the RCS and is done shortly after the RCS has been depressurized below 1000 psig. The redundant pressure and level alarms on each accumulator would remind the operator to close these valves, if any were inadvertently left open. Power is disconnected at the motor control center after the valves are closed. In the event that the operator is unable to close any o
f
these valves, the accumulator vent valve is opened to depressurize the accumulator and avoid the addition of excess water inventory into the RCS.

During plant startup, the operator is instructed, via procedures, to energize and open these valves before the RCS pressure exceeds 1000 psig. Monitor lights in conjunction with an audible alarm will alert the operator should any of these valves be left inadvertently closed once the RCS pressure increases beyond the safety injection unblock setpoint. After these valves have been opened, power to these valves is disconnected at the motor control center.

The accumulator isolation valves are not required to move during power operation or in a post-accident situation, except for valve testing. For a discussion of limiting conditions

CALLAWAY - SP

TABLE 6.3-1 (Sheet 2)

| | |
|---|---|
| Safety Injection Pumps | |
| Number | 2 |
| Design pressure, psig | 1,750 |
| Design temperature, °F | 300 |
| Design flow rate, gpm | 425 |
| Design head, ft | 2,680 |
| Maximum flow rate, gpm | |
| Injection phase | 675[a] |
| Recirculation phase | 691[a] |
| Head at maximum flow rate, ft | 1,650 |
| Discharge head at shutoff, ft | 3,645 |
| Required NPSH at max flow, ft | 17 |
| Available NPSH, ft | 43.8 |
| Design code | ASME III, Class 2 |
| Seismic design | Category I |
| Driver: | |
| Type | Electric motor |
| Horsepower, hp | 450 |
| Rpm | 3,600 |
| Power | 4,160 V, 60 Hz, 3-phase, Class 1E |
| Start time | ≤5 sec |
| Design code | NEMA |
| Seismic design | Category I |
| Residual Heat Removal Pumps | |
| Number | 2 |
| Design pressure, psig | 600 |
| Design temperature, °F | 400 |
| Design flow, gpm | 3,800 |
| Design head, ft | 350 |
| NPSH required at 4,800 gpm, ft | ~~21.7~~ 21.8[b] |
| Available NPSH at 4,800 gpm, ft | ~~25.7~~ 26.2[b] |
| Design code | ASME III, Class 2 |
| Seismic design | Category I |
| Driver: | |
| Type | Electric motor |
| Horsepower, hp | 500 |
| Rpm | 1,800 |
| Power | 4,160 V, 60 Hz, 3-phase, Class 1E |
| Start time | ≤5 sec |
| Design code | NEMA |
| Seismic design | Category I |
| Total Sump Screen Head Loss (max.), ft | 4.2[b] |
| NPSH Margin, ft | 0.2[b] |

(a) Includes miniflow (30 gpm)

(b) Values from MP 15-0017.

Table 6.3-13

Containment Recirculation Sump Debris Limits

| Debris Type | Analyzed Source Limit |
|---|---|
| Low-density fiberglass | 300 lbm (includes 50-lbm margin above current plant configuration) |
| Particulates (failed coatings) | 25.1 $ft^3$ |
| Latent dirt and dust | 170 lbm |
| Aluminum Oxyhydroxide chemical precipitate | 474.0 lbm (215 kg) |
| Calcium Phosphate chemical precipitate | 55.1 lbm (25 kg) |
| Miscellaneous debris blockage | 200 $ft^2$ |

# APPENDIX 6.3A - RESOLUTION OF NRC GENERIC LETTER 2004-02

## 6.3A.1 INTRODUCTION AND SUMMARY

NRC Generic Letter 2004-02 (GL 2004-02),"Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," requires licensees to perform an evaluation of the emergency core cooling system (ECCS) and containment spray system (CSS) emergency recirculation functions, and the flowpaths necessary to support those functions, based on the potential susceptibility of sump screens to debris blockage during design basis accidents requiring recirculation operation of ECCS or CSS. GL 2004-02 resulted from Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance." As a result of the evaluation required by GL 2004-02 and to ensure system function, sump design modifications were implemented.

The plant licensing basis considers long-term core cooling following a LOCA as identified in 10 CFR 50.46. Long-term cooling is supported by the ECCS, which includes the ECCS centrifugal charging pumps (CCPs) and the safety injection (SI), and residual heat removal (RHR) systems. The CCPs, SI system, and RHR system, as well as the CSS, are subject to the effects of LOCA debris because they rely on the containment recirculation sumps in the recirculation mode. Debris from non-LOCA events (steam line breaks) is not in the scope of the GL 2004-02 evaluation because those events do not result in ECCS or CSS operation in the recirculation mode where debris would become a factor. GL 2004-02 sump performance evaluation activities include the following:

- Containment walkdowns to identify and quantify sources of debris
- Debris generation and transport analysis
- Calculation of required and available net positive suction head (NPSH) for ECC and CS pumps
- Containment recirculation sump strainer performance requirements
- Containment recirculation sump strainer structural analyses
- Operations procedure changes
- Debris effects downstream of the strainers and sumps, including effect on core flow and of wear and erosion on internal components
- Debris effects upstream of the strainers and sumps that might impede water returning to the recirculation pool
- Chemical effects associated with debris and with metal corrosion
- Plant-specific testing of strainer loads and chemical product formation potential
- Risk-informed evaluation of debris effects not bounded by plant-specific strainer testing

A deterministic evaluation supplemented by a risk-informed evaluation was performed to respond to GL 2004-02. The combined evaluation, identified as "RoverD," provides confidence that the sump design supports long-term core cooling following a design basis loss of coolant accident. The RoverD approach includes a model/methodology (CASA Grande) used to evaluate the magnitude of pipe breaks (LOCAs) required to exceed the risk-based metric referred to as the fiber threshold (for which the results of head-loss testing were taken into account). The risk-informed evaluation meets the acceptance guidelines for a “very small” change in risk as defined in Regulatory Guide 1.174.

To complement the RoverD analysis, an analysis based on WCAP-16406, “Evaluation of Downstream Sump Debris Effects in Support of GSI-191 (PA-SEE-0195),” was performed to address ex-vessel downstream effects. An analysis based on WCAP-17788, “Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090),” was also performed to address in-vessel downstream effects.

The licensing basis with regard to effects of debris is that there is a high probability that ECCS and CSS can perform their design basis functions based on successful plant-specific prototypical strainer testing using deterministic NRC-approved assumptions, and that the risk from breaks that could generate debris that is not bounded by the testing is very small and is acceptable in accordance within the criteria of Regulatory Guide 1.174.

The use of a risk-informed method to assess the effects of breaks that exceed the bounds of strainer testing, rather than the deterministic methods prescribed in the regulations, required exemptions to 10 CFR 50.46(a)(i), and the applicable General Design Criteria of 10 CFR 50 Appendix A, i.e., General Design Criterion (GDC) 35, GDC 38, and GDC 41, which have been granted pursuant to 10 CFR 50.12.

### 6.3A.1.1 Deterministic Element

The deterministic element of the combined deterministic and risk-informed evaluation applies Callaway plant-specific strainer testing performed using accepted guidance to establish an upper limit on the amount of LOCA-generated debris that can be handled successfully by the ECCS strainers and also satisfy core blockage limits.

The containment condition assessments included the identification of miscellaneous solid objects such as labels and tags. Approximately 330 $ft^2$ of legacy tags, labels and tape used for equipment identification remain in containment, of which approximately 246 $ft^2$ are metal equipment tags that are located within a zone-of-influence ZOI for an analyzed break location. Plant geometry and strainer testing results have confirmed that these metal equipment tags, and a limited quantity of metal tags used to mark radiological survey locations for trending, are not transportable to the sumps by recirculation flow. The remaining quantity of transportable miscellaneous debris (approximately 84 $ft^2$) is bounded by the assumption of 200 $ft^2$ that was used to establish the strainer testing parameters. Specifically, allowing for an assumed 75% overlap of transported miscellaneous debris items, the effective strainer area was decreased by 150 $ft^2$ during testing to account for any remaining miscellaneous debris of this type.

Debris Amounts Used for the Test

Plant debris amounts that were scaled testing performed in June 2016 are given in Table 6.3A-2. These values represent the debris that gets transported to the strainer modules of one operating train of strainers.

The full 30-day chemical precipitate load is assumed to arrive at the strainer at the earliest possible time with no credit for settling or nucleation on containment surfaces. The quantity of precipitate arriving at the strainer surface is expected to be significantly lower than the calculated or tested amounts. In addition, the precipitate would be expected to arrive gradually and resultant head loss would be compensated by increasing head loss margins as the containment pool cools.

The sump temperature used for chemical corrosion is from the containment LOCA pressure-temperature analysis that maximizes the sump temperature by using the maximum temperatures for cooling water to the heat exchangers and for the water of the ultimate heat sink and uses very conservative mass and energy release rates from the reactor. The maximum sump temperature is assumed to persist for 30 days for the purpose of overestimating chemical corrosion.

The containment water level was determined using conservative input values for the pool contributions and conservatively accounting for items such as holdup in some locations in the containment, filling of empty pipe, water in transit, steam holdup, etc. The minimum strainer submergence used for the NPSH evaluation was 0.1 ft for SBLOCA and 0.6 ft for LBLOCA, which occurs at RHR switchover before a full debris bed is formed and is conservative compared to the calculated minimum available submergence of 2 inches for SBLOCA and 8 inches for LBLOCA.

Higher-than-expected pump flow rates corresponding to maximum documented licensing values are conservatively used for the NPSH evaluation and during strainer head loss testing. Flow rates for the expected LOCA condition would generally be lower than as-tested flow rates and operator actions to further reduce flow would be expected following indications of significant strainer head loss. Reduced flow reduces debris-induced head loss.

For the NPSH evaluation, all of the debris is assumed to form a debris bed on the strainer at the start of recirculation. The debris consists of all of the insulation fiber fines, all of the coating particulates, and all of the chemical precipitates that were present on the strainer at the end of the full debris load strainer test.

A clean strainer head loss is applied in plant NPSH calculations for the RHR and CSS pumps that ranges between 0.2 and 0.7 feet of water depending on whether the flow rate is for RHR maximum flow alone or for both RHR and CSS maximum flows combined, and on water temperature of either 212°F or 140°F. Clean strainer head loss is also accounted for when comparing NPSH available to strainer testing results.

Containment accident pressure of 1.7 psi is credited for available NPSH during the LBLOCA phase when containment temperature is above 212°F to assure no flashing to steam occurs in the debris bed (approximately 10% of available containment pressure). In general, for a sump temperature of 212°F and higher, the NPSH available calculation assumes that the containment pressure is equal to the vapor pressure. For sump temperatures lower than 212°F, the containment pressure is assumed to be equal to atmospheric pressure of 14.7 psia, as if there is loss of containment. Overpressure credit is not needed to meet the NPSH, air release, or strainer buckling strainer performance criteria.

The maximum head loss experienced under full load debris testing of one strainer at minimum assumed pool temperature does not exceed the mechanical buckling/crush evaluation of the Callaway containment recirculation sump strainers.

The Callaway containment recirculation sump strainers were structurally qualified with concurrent seismic loads for a debris loading of 4330.8 lbm, which exceeds the maximum calculated debris load of 4303 lbm transported among all break scenarios that pass the RoverD fiber threshold, when crediting Callaway's 2018 reclassification of qualified coatings systems, based on fall 2017 outage surveys.

Break scenarios are conservatively assigned to core damage in the RoverD approach when the strainer structural qualification limit as determined from the Code allowable stress value is exceeded.

Tests performed to support the WCAP-17788 analysis that were specific to Callaway LOCA accident conditions (but with a conservatively high pH that induces more aluminum corrosion) show no concern for chemical product formation in the core prior to switch over to hot-leg injection.

### 6.3A.1.2 Risk Informed Element - RoverD Summary

As noted in Section 6.3A.1, the approach used to close GL 2004-02 includes a deterministic element and a risk-informed element (risk over deterministic, or "RoverD"). The effects of debris that are bounded by the plant-specific testing are deterministically mitigated in accordance with NRC-accepted methodology for resolution of GL 2004-02. Section 6.3A.1.1 describes the deterministic evaluation.

The risk-informed element identifies LOCA break sizes that may produce more than the deterministically tested amount of fine fiber debris (300 lbm). Breaks that may generate and transport fine fiber debris in excess of the tested amount are conservatively assumed to cause core damage. The break frequency for the smallest break size (i.e., highest frequency for pipe breaking) at each weld that can generate and transport an amount of fine fiber debris that exceeds the amount that was tested is added to the total ΔCDF in the risk-informed evaluation. Sixty (60) welds were identified at Callaway that can exceed the tested fiber limit and contribute to ΔCDF. The geometric mean pipe break exceedance frequency from NUREG-1829 is used to evaluate total ΔCDF from all non-isolable weld breaks that can transport more than 300 lb of fine fiber.

The risk-informed element also evaluates in-vessel effects to confirm that there are no failures from in-vessel effects for the scope of LOCAs that are satisfactorily addressed in the deterministic element. Material that passes through the ECCS strainer debris bed was evaluated for the possibility of accumulation in fuel channels, which can impede cooling flow. WCAP-17788 analysis was used to confirm that fiber accumulation in the core for the RoverD limit of 300 lbm of transported fiber mass does not exceed WCAP-17788 guidelines for grams of fiber per fuel assembly for either hot-leg or cold-leg LOCA. Break scenarios are conservatively assigned to core damage in the RoverD approach when estimates of total transported fiber exceed 300 lbm.

#### 6.3A.1.2.1 RoverD risk quantification process summary

RoverD involves the following steps to assess the risk associated with the effects of LOCA debris:

1. A plant-specific test is performed that follows accepted protocols (deterministic element of RoverD). The tested debris composition, and in particular the fiber quantity, establishes deterministic strainer performance limits.
   - The amount of fiber fines tested (300 lbm) as well as the sump configuration (one of two ECCS trains) are noted. The plant configuration is important to ensure that the test bounds other plant states. Testing a single strainer with the full debris load creates the largest debris-induced head loss. Fine fiber is used in the test, because it is the most transportable form of the low-density fiberglass (LDFG) debris created in the break scenario.
   - The test results are applied to all strainer performance criteria to ensure they are met using deterministic analysis requirements (i.e., vortexing, structural margin, flashing, NPSH, and air evolution)
2. In-vessel performance criteria (core cooling, including fiber effects, boric acid precipitation) must be met under the conditions tested.
   - WCAP-17788 methodology is used to confirm that when the strainer receives 300 lbm of fine fiber, the fiber deposited in fuel channels for a hot-leg break (80.22 g/FA analyzed) does not exceed the threshold value from Section 6.5.5.10.c of WCAP 17788-P, Volume 1, Rev. 0.
   - WCAP-17788 methodology is used to confirm that when the strainer receives 300 lbm of fine fiber, the fiber deposited in the fuel channels for a cold-leg break (5.5 g/FA analyzed) does not exceed the threshold value from Section 9.2.1 of WCAP-17788-P, Volume 1, Rev. 0.
   - For boric acid precipitation, the generic WCAP-17788 evaluation shows that hot leg switchover timing is appropriate with debris effects considered.

3. All postulated break locations, break sizes, and the corresponding amounts of LDFG fines transported to the strainers (including erosion and latent fiber) are itemized and noted.

   - 50 lbm of transported fiber is added to every break scenario to provide explicit operational margin for future discovery.

   - 133 lbm of transported particulate is added to every break scenario to provide explicit operational margin for future discovery or reclassification of coatings condition. Known particulate testing margin of 227 lbm raises total particulate margin to 360 lbm.

4. The amount of transported fiber fines and the amount of transported particulate in each break scenario is compared to the tested amounts.

   - If the analyzed amounts (including margins) are equal to or less than the tested amounts, categorize the scenario as 'deterministic' with no associated risk.

   - If the analyzed amounts (including margin) exceed the tested amounts, categorize the scenario as 'risk-informed' and add the scenario frequency to total ΔCDF.

5. The risk contribution (including in-vessel considerations, which do not add risk for Callaway) of scenarios in the risk-informed category are evaluated against the Regulatory Guide 1.174 quantitative criteria for {CDF, ΔCDF}, (read as total CDF and incremental change in CDF caused by debris) and {LERF, ΔLERF}, (read as total LERF, and incremental change in LERF caused by debris).

   - The 25-yr PWR geometric mean exceedance frequency from NUREG-1829 is used to add incremental ΔCDF for the smallest size break at each weld that can generate and transport fine fiber debris in excess of what was tested (only 60 weld breaks are capable of generating and transporting more than 300 lbm of fiber).

   - The calculated {CDF, ΔCDF} values are checked against the quantitative requirement of Regulatory Guide 1.174, Region III, noting that total CDF is obtained from the PRA model of record.

   - ΔLERF is calculated by assuming that (1) total LERF is 10 times lower than total CDF (standard industry ratio), and (2) the proportional increase in LERF is the same as the proportional increase in CDF (i.e., ΔLERF divided by LERF) equals ΔCDF divided by CDF). These assumptions mean that calculated ΔCDF is multiplied by 0.1 to get calculated ΔLERF.

- Calculated {LERF. ΔLERF} values are checked against the quantitative requirement of Regulatory Guide 1.174, Region III. Total LERF is obtained from the PRA model of record is bounded by the common industry assumption that LERF is 10 times lower than total CDF.

- Other requirements of Regulatory Guide 1.174 (e.g., safety margin, defense in depth) are verified to be met.

6. If all requirements are met for the risk-informed category, the performance is deemed acceptable.

6.3A.1.2.2 Reactor Containment Building Debris Generation and Transport

Debris Generation

A break size and location define a scenario from which is derived the amount of fiber fines that arrive in the containment recirculation sumps. All breaks are assumed to occur at high-energy line welds at or inside the first isolation valve. Locations of plant equipment and structures and potential fiber insulation and qualified coatings targets are identified from a CAD model. A 17-times pipe diameter (17D) zone-of-influence (ZOI) at each RCS weld location is used to quantify the amount of fine fiber generated. Each scenario-specific break is represented in the simulation by either a spherical ZOI for double-ended guillotine breaks or by a hemispherical ZOI for partial breaks. The computer evaluation varied the orientation of the break location around the circumference of the weld locations to assure that the maximum debris generation was attained. Credit is taken for shielding by concrete walls, by the reactor vessel, and by the pressurizer skirt. Guidance prohibits taking credit for shielding by steam generators, reactor coolant pumps, and other large equipment in containment.

Debris Transport

A transport logic tree is used to quantify the amount of fiber transported to the containment pool. The bounding transport fractions applied to various debris types and sizes for all breaks are selected from detailed analyses of representative large breaks occurring in the upper and lower Steam Generator compartment and outside the bioshield wall, so that transport fractions for all other possible break locations in the containment building are estimated conservatively. Guidance allows some credit for debris retention on gratings, so spatial break location is important to the debris transport analysis.

In addition to the fiber fines generated directly in the ZOI, fines are also created from erosion of large and small pieces of insulation impinged by spray and when settled in the recirculation pool. Finally, 15% of latent debris (resident dust and dirt) is assumed to be fine fiber and it is also transported to the containment pool with the eroded fiber fines.

### 6.3A.1.2.3 LOCA frequencies and results

Determination of Core Damage Frequency

Sixty (60) weld locations are identified on the pressurizer surge line, RHR, and RCS main loop piping where a sufficient amount of fiber debris can be generated and transported to the sumps to exceed the amount of fine fiber debris included in Callaway plant-specific strainer testing described in Section 6.3A.1.1. To provide break size perspective, that scope is generally described as breaks larger than approximately 9" ID in those locations. No explicit credit is taken for the smaller likelihood of single-train operation compared to the highly reliable, expected, two-train accident response.

Weld locations in the risk-informed category are listed in Table 6.3A-1.

A continuum break model is used that assumes small breaks can occur on large pipes. The model evaluates the total frequency of all non-isolable weld breaks that can transport more than 300 lb of fine fiber by using the geometric mean aggregation of NUREG-1829-elicited 25-year PWR LOCA exceedance frequencies in order to confirm a ΔCDF frequency of less than 1.0E-06/yr (in RG 1.174 Risk Region III). The corresponding ΔLERF is less than 1.0E-07/yr (also in RG 1.174 Risk Region III).

The risk-informed evaluation meets the RG 1.174 guidance with respect to defense in depth in that the following aspects of the facility design and operation are practiced, maintained, monitored, and remain otherwise unaffected:

- Functional requirements and the design configuration of systems
- Existing plant barriers to the release of fission products
- Design provisions for redundancy, diversity, and independence
- Plant's response to transients or other initiating events
- Preventive and mitigative capabilities of plant design features
- Configuration management and plant cleanliness practices

The Emergency Operating Procedure (EOP) framework has guidance for monitoring for the loss of containment recirculation sump recirculation capabilities and actions to be taken if this condition occurs. Actions are summarized below:

(1) Monitoring for proper pump operation, core exit thermocouples, and reactor water level indication,

(2) Refilling the RWST for injection flow,

(3) Using injection flow from alternate sources, and

(4) Transferring to combined hot leg/cold leg injection flow paths.

The evaluation demonstrates sufficient safety margin, as summarized below.

- Pressurizer surge, safety and relief nozzle welds susceptible to PWSCC were mitigated by pre-emptive full structural weld overlays in the 2007 spring refueling outage.

- Water-jet peening was performed on the reactor nozzle welds in the 2016 spring refueling outage.

- The debris generation analysis does not take credit for shielding within the ZOI by equipment (e.g. steam generators, reactor coolant pumps) and large piping. (While conservative for most LOCA scenarios, this assumption conforms to regulatory guidance and do not provide quantifiable safety margin.)

- Instantaneous failure and transport of 100% of the unqualified coatings inside containment as particulates is a very conservative assumption. (While conservative, this assumption conforms to regulatory guidance and does not provide quantifiable safety margin.)

- Chemical effects were conservatively tested by assuming maximum recirculation pool temperature persists for 30 days. (While the following assumptions are conservative, they conform to regulatory guidance and do not provide quantifiable safety margin.)

    - The full 30-day chemical precipitate load is assumed to arrive at the strainer at the earliest possible time with no credit for settling or nucleation on containment surfaces. The effect of the precipitate is reduced at the realistic arrival time because of the lower decay heat load and containment pool temperature.

    - Based on other plant-specific testing, the quantity of precipitate arriving at the strainer surface is expected to be significantly lower than the calculated amounts.

    - The precipitate is expected to arrive gradually and resultant head loss would be compensated by increased head loss margins.

- The available NPSH is conservatively determined. (While conservative, these assumptions conform to regulatory guidance and do not provide quantitative safety margin.)

    - The sump temperature is maximized from the containment LOCA pressure-temperature analysis.

- The containment pool water level assumption is conservatively low.
- No credit is given for containment accident pressure. (A small containment overpressure is needed to ensure that boiling does not occur within the debris bed (i.e., to prevent disruption of the bed by boiling) at maximum pool temperature, but is not credited for NPSH).

- Fifty (50) pounds of extra transported fiber are added to every break scenario so that more break scenarios exceed the tested limit and contribute to a conservative estimate of risk. The extra fiber represents explicit safety margin that can be assigned to future discovery of adverse containment conditions without increasing the present ΔCDF and ΔLERF estimates.

- One hundred and thirty three pounds (133 lbm) of extra transported particulate are added to every analyzed break scenario so that more break scenarios exceed the tested limits and contribute to a conservative estimate of risk. In addition, the tested particulate load exceeds the largest DEGB total particulate transport by 227 lbm, giving a total margin of 360 lbm. The extra particulate represents explicit safety margin that can be assigned to future discovery of additional debris sources in containment without increasing the present ΔCDF and ΔLERF estimates.

- No credit is taken in the risk-informed analysis for the smaller likelihood of single-train ECCS response compared to the reliable and expected two-train accident response. Careful quantification of the single-train configuration likelihood could be applied as a direct reduction factor to the present ΔCDF and ΔLERF estimates.

WCAP-17788 evaluations show that there is safety margin remaining in potential in-vessel debris effects with respect to both hot-leg and cold-leg break configurations (see Section 6.3A.1.2.1).

### 6.3A.1.3 Exemptions to Regulations

In support of the Callaway risk-informed approach to addressing GSI-191 and response to GL 2004-02, Callaway was granted exemptions under 10 CFR 50.12 from certain requirements in 10 CFR 50.46 and 10 CFR 50 Appendix A General Design Criteria.

The specific exemptions pertain to requirements for deterministic analysis of ECCS and CSS functions for core cooling, and containment heat removal and atmosphere cleanup following a postulated loss of cooling accident and affect the following requirements:

- 10 CFR 50.46(a)(1) - the governing requirement in 10 CFR 50.46 to establish General Design Criterion (GDC) 35 as the technical design basis for ECCS analysis.

- GDC 35, Emergency Core Cooling

- GDC 38, Containment Heat Removal
- GDC 41, Containment Atmosphere Cleanup

The exemptions allow use of the risk-informed methodology described in this appendix to account for the probabilities and uncertainties associated with mitigation of the effects of debris following postulated LOCAs instead of using the deterministic analyses required by the regulation or General Design Criteria.

The scope of the exemptions applies for all debris effects addressed in the risk-informed element of the RoverD methodology described in this appendix, which is associated with LOCA break sizes and locations that potentially generate fine fiber debris that exceeds the quantity bounded by plant-specific testing described in Section 6.3A.1.1. That scope is generally described as breaks larger than approximately 9" ID in locations where a sufficient amount of fiber debris can be generated and transported to the sumps to exceed the amount of fine fiber debris in the plant-specific testing described in Section 6.3A.1.1.

The key elements of each of the exemption requests are:

1. The exemption applies only to the effects of debris as described in this Appendix 6.3A.

2. The exemption applies to any LOCA break that can generate and transport fiber debris that is not bounded by Callaway plant-specific testing, provided that the ΔCDF and ΔLERF associated with the break size remains in Region III of RG 1.174 (see Table 6.3A-1).

Section 3.1.6 provides additional information on Callaway's compliance with GDC 35, 38, and 41.

### 6.3A.1.4 Technical Specifications

The Technical Specifications for the ECCS were revised to move the sump surveillance requirement for inspection of the containment sumps into new TS 3.6.8, “Containment Sumps”. The operability requirements specified per the Limiting Condition for Operation (LCO) of TS 3.6.8 are based in part on meeting the analyzed containment accident generated and transported debris limits referred to in the TS, as determined from the RoverD analysis. Condition A under the LCO is entered when one or more containment sumps is declared inoperable due to containment accident generated and transported debris exceeding the analyzed limits. The two sumps are considered part of a single support system because containment accident generated and transported debris issues that could render one sump inoperable could render all of the sumps inoperable. The required Completion Time for restoring compliance (Operability) is 90 days, provided certain other Required Actions for Condition A are met. The Completion Time is based on the low probability of an initiating event and the very low risk from the effects of debris, as demonstrated in the RoverD evaluation.

Additional information is provided in the Technical Specification Bases for the ECCS and CSS.

The discovery of debris or a debris source(s) in the containment not accounted for in analysis would involve evaluation of the quantity, nature and transportability of the debris in question to determine if it is within the Callaway debris analysis. For Operability determinations performed in such instances, the application of probabilistic risk is not allowed (in keeping with industry guidance), but credit can be taken for fiber and particulate safety margins inherent to the analysis. (See safety margin discussion in Section 6.3A.1.2.3.)

## 6.3A.2 CHANGE CONTROL AND REPORTING

This section describes those methodologies or elements of the methodologies used in support of the RoverD approach for addressing sump debris concerns such that changes to the identified methodologies or elements may invlove additional regulatory requirements for NRC approval or notification.

### 6.3A.2.1 Change Control for Methods of Evaluation

Changes to any of the key methodologies (including acceptance criteria) used in the RoverD approach for addressing GSI-191/GL 2004-02 concerns, as described in Section 6.3A.1 of this Appendix, are to be evaluated as a potential "departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses," consistent with 10 CFR 50.59(c)(2)(viii) (notwithstanding the fact that some of the methodologies are risk-informed or involve risk-informed elements).

The key methods of evaluation within the RoverD methodology subject to change control are the following:

1. The methodology for performing integrated calculations for overall risk evaluation or to identify changes in overall risk (CASA Grande).

2. The methodology for quantifying the pipe break frequencies used to calculate the change in CDF and LERF.

   a. The pipe break frequency source (NUREG-1829).

   b. The methodology for calculation of Di(small), the smallest break at each weld capable of exceeding tested debris limits.

   c. The methodology used to identify break locations.

   d. The methodology used to interpolate pipe break exceedance frequencies to quantify the ΔCDF associated with pipe breaks

   e. The methodology used to calculate ΔLERF for effects of debris

f. Shifts in absolute CDF or absolute LERF causing ΔCDF or ΔLERF to exceed RG 1.174 risk region III limits

g. Changes in ΔCDF or ΔLERF exceeding limits of RG 1.174 risk Region III

3. The assumption that fine fiber is to be applied as the governing debris source.

   a. The methodology used to quantify the amount of fiber generated at each break location, including assumed ZOI. This requirement applies to the criteria, but not to the tool or program used; i.e., programs other than CASA Grande may be used.

   b. The use of the June 2016 Callaway-specific testing to establish the deterministic baseline for the quantity of fine fiber.

4. The assumptions and methods in the WCAP-17788 in-vessel effects analyses.

5. The availability of key sources of defense-in-depth.

   a. Capability for containment heat removal unaffected by debris on ECCS strainers (Containment Air Coolers).

   b. Capability to refill the RWST under existing Emergency Operating Procedures (EOPs).

6. The assumptions and methods for performing ex-vessel downstream effects evaluations.

7. The assumptions and methodology for debris transport in containment, as described in Section 6.3A.1.2.2.

8. The methods used to estimate generation of debris types other than fiber (e.g., chemical precipitates, coatings, or other potential debris sources)

9. Limits on other debris types, as specified in Table 6.3A-2.

### 6.3A.2.2 Changes to Plant Design and Operating Practices

In addition to the controls of Section 6.3A.2.1, changes to plant design and operating practices shall be subject to the following requirements:

1. Procedural controls establish limits on the introduction of new debris sources into containment, particularly fine fiber, chemical precipitate or particulate sources, to assure the deterministic licensing basis testing and calculations performed for the RoverD methodology remain bounding.

2. Every 4.5 years (three 18-month refueling cycles), a review of the validity of the risk-informed LAR will be performed that considers industry-wide changes in regulatory guidance, initiating event frequencies, component and equipment

reliability factors, PRA model changes that pertain to the risk-informed GSI-191 resolution, and other emergent industry and plant-specific concerns. The review will include an assessment of cumulative containment configuration changes and changes to operating practices (both beneficial and detrimental) that occur during intervening outages.

### 6.3A.2.3 Reporting

Nonconforming conditions that make the strainer(s) inoperable (during the Modes of applicability) for longer than the required TS completion time will meet the 10 CFR 50.73 reporting criteria for a condition prohibited by TS. Conditions that cause the containment recirculation sump strainers to be inoperable and result in the debris-related ΔCDF or ΔLERF to be greater than the RG 1.174 Region III acceptance guidance are to be reported in accordance with 10 CFR 50.72 and 10 CFR 50.73 as unanalyzed conditions that significantly degrade plant safety. Conditions that cause emergency sump strainers to be inoperable must exceed the RG 1.174 Risk Region III – low risk threshold.

## 6.3A.3 REFERENCES

1. Callaway Operating License Amendment No. 2xx dated Xxxx xx, 20xx.

2. WCAP-16406-P-A, “Evaluation of Downstream Sump Debris Effects in Support of GSI-191 (PA-SEE-0195)," Rev. 0, August 2007

3. WCAP-17788-P, “Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090),” Rev. 0, July 2015

TABLE 6.3A-1 (Sheet 1 of 3)

WELD LOCATIONS IN THE RISK-INFORMED CATEGORY

| # | Weld Location Name | Smallest Break Size to Fail (inches) | Fiber Transported at Smallest Break Size (lbm) |
|---|---|---|---|
| 1 | WELD EBB01B-RSG-OUTLET-SC010 | 11.855 | 300.091 |
| 2 | WELD 2-BB-01-3065B-WDC-002-FW2 | 11.795 | 300.354 |
| 3 | WELD 2-BB-01-F206 | 11.495 | 300.096 |
| 4 | WELD 2-BB-01-S204-3 | 11.885 | 300.149 |
| 5 | WELD 2-BB-01-F208 | 11.565 | 300.156 |
| 6 | WELD 2-BB-01-S205-4 | 11.125 | 300.025 |
| 7 | WELD 2-BB-01-F207 | 11.525 | 300.048 |
| 8 | WELD 2-BB-01-S201-2 | 27.500 | 313.423 |
| 9 | WELD 2-BB-01-F201 | 10.855 | 300.254 |
| 10 | WELD 2-BB-01-S202-2 | 13.275 | 300.183 |
| 11 | WELD 2-BB-01-3065B-WDC-001-FW1 | 13.225 | 300.217 |
| 12 | WELD EBB01B-RSG-INLET-SC010 | 13.255 | 300.190 |
| 13 | WELD 2-EJ-04-FW9 | 10.500 | 327.631 |
| 14 | WELD 2-EJ-04-S018-C | 10.500 | 329.261 |
| 15 | WELD 2-EJ-04-FW8 | 10.500 | 326.983 |
| 16 | WELD 2-EJ-04-S018-E | 10.500 | 351.002 |
| 17 | WELD 2-EJ-04-F031 | 9.985 | 300.095 |
| 18 | WELD EBB01A-RSG-OUTLET-SC010 | 10.905 | 300.032 |
| 19 | WELD 2-BB-01-3065A-WDC-002-FW2 | 10.805 | 300.247 |
| 20 | WELD 2-BB-01-F106 | 9.965 | 300.022 |

TABLE 6.3A-1 (Sheet 2 of 3)

WELD LOCATIONS IN THE RISK-INFORMED CATEGORY

| # | Weld Location Name | Smallest Break Size to Fail (inches) | Fiber Transported at Smallest Break Size (lbm) |
|---|---|---|---|
| 21 | WELD 2-BB-01-S104-3 | 9.915 | 300.010 |
| 22 | WELD 2-BB-01-F108 | 9.915 | 300.054 |
| 23 | WELD 2-BB-01-S105-4 | 9.145 | 300.362 |
| 24 | WELD 2-BB-01-F107 | 9.595 | 300.226 |
| 25 | WELD 2-BB-01-S102-2 | 11.755 | 300.091 |
| 26 | WELD 2-BB-01-3065A-WDC-001-FW1 | 11.525 | 300.508 |
| 27 | WELD EBB01A-RSG-INLET-SC010 | 11.565 | 300.063 |
| 28 | WELD 2-RV-302-121-A | 27.325 | 300.076 |
| 29 | WELD 2-BB-01-F102 | 26.975 | 300.068 |
| 30 | WELD 2-BB-01-S101-2 | 27.500 | 345.081 |
| 31 | WELD 2-BB-01-F101 | 9.475 | 300.219 |
| 32 | WELD EBB01C-RSG-OUTLET-SC010 | 11.615 | 300.129 |
| 33 | WELD 2-BB-01-3065C-WDC-002-FW2 | 11.515 | 300.400 |
| 34 | WELD 2-BB-01-F306 | 10.945 | 300.219 |
| 35 | WELD 2-BB-01-S304-3 | 11.295 | 300.226 |
| 36 | WELD 2-BB-01-F308 | 11.255 | 300.364 |
| 37 | WELD 2-BB-01-S305-4 | 10.585 | 300.050 |
| 38 | WELD 2-BB-01-F307 | 11.165 | 300.014 |
| 39 | WELD 2-BB-01-F301 | 11.415 | 300.700 |
| 40 | WELD 2-BB-01-S302-2 | 13.405 | 300.053 |

TABLE 6.3A-1 (Sheet 3 of 3)

WELD LOCATIONS IN THE RISK-INFORMED CATEGORY

| # | Weld Location Name | Smallest Break Size to Fail (inches) | Fiber Transported at Smallest Break Size (lbm) |
|---|---|---|---|
| 41 | WELD 2-BB-01-3065C-WDC-001-FW1 | 12.995 | 300.271 |
| 42 | WELD EBB01C-RSG-INLET-SC010 | 13.015 | 300.404 |
| 43 | WELD 2-BB-01-S402-2 | 12.385 | 300.334 |
| 44 | WELD 2-BB-01-3065D-WDC-001-FW1 | 11.825 | 300.020 |
| 45 | WELD EBB01D-RSG-INLET-SC010 | 11.875 | 300.188 |
| 46 | WELD 2-BB-01-F401 | 9.905 | 300.507 |
| 47 | WELD EBB01D-RSG-OUTLET-SC010 | 10.745 | 300.278 |
| 48 | WELD 2-BB-01-3065D-WDC-002-FW2 | 10.635 | 300.075 |
| 49 | WELD 2-BB-01-F406 | 9.845 | 300.122 |
| 50 | WELD 2-BB-01-S404-3 | 9.695 | 300.179 |
| 51 | WELD 2-BB-01-F408 | 9.605 | 300.489 |
| 52 | WELD 2-BB-01-S405-4 | 9.195 | 300.327 |
| 53 | WELD 2-BB-01-F407 | 9.635 | 300.222 |
| 54 | WELD 2-EJ-04-S016-B | 10.500 | 310.438 |
| 55 | WELD 2-EJ-04-S016-C | 10.500 | 318.508 |
| 56 | WELD 2-EJ-04-S016-D | 10.500 | 316.902 |
| 57 | WELD 2-EJ-04-S016-E | 10.330 | 300.053 |
| 58 | WELD 2-EJ-04-S016-G | 9.485 | 300.239 |
| 59 | WELD 2-EJ-04-S016-H | 10.500 | 403.032 |
| 60 | WELD 2-EJ-04-F025 | 10.155 | 300.192 |

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TABLE 6.3A-2

DEBRIS AMOUNTS FOR JUNE 2016 FULL DEBRIS LOAD TEST (AT PLANT SCALE)

| Debris Item | Debris Load | Test Surrogate Material |
|---|---|---|
| | | |
| **Fiber** | | |
| NUKON low-density fiberglass (LDFG) | 125 $ft^3$ (300 lbm)<br>(includes 30 lbm of latent fiber and LOCA-damaged insulation) | Heat-treated NUKON<br>(100% Fines) |
| | | |
| **Particulates** | | |
| Latent particulate, dust, dirt | 30 lbm | Dirt mixture |
| ZOI-Damaged Coatings + Unqualified Coatings<br>(includes Acrylic, Epoxy, and Inorganic Zinc) | 5770 lbm | Silicate mixture |
| Total Particulate | 5800 lbm | |
| | | |
| Particulate Notes:<br>1. Tested particulate mass (5800 lbm) exceeds largest DEGB total particulate transport (5,573 lbm) by 227 lbm<br>2. Additional analytic margin of 133 lbmin the risk analysis gives a total particulate margin of 360 lbm<br>3. The volume of debris on the strainer displaces flow and causes head loss, but mass provides a more concise summary when densities vary.<br>4. Latent particulate of 30 lbm is based on Callaway plant walkdown | | |
| | | |
| **Chemical Precipitates** | | |
| Sodium Aluminum Silicate | 474 lbm | WCAP Aluminum Oxyhydroxide (AlOOH)<br>(acceptable surrogate for Sodium Aluminum Silicate) |
| Aluminum Oxyhydroxide | 0 lbm<br>(Callaway pH and temperature do not produce AlOOH) | |
| Calcium Phosphate | 55 lbm | WCAP Calcium Phosphate ($Ca_3(PO_4)_2$) |
| Total Chemical Product | 529 lbm | |
| | | |
| **Miscellaneous Debris** | | |
| Tags, tape, labels | 200 $ft^2$ | Test velocity increased to account for loss of flow area |

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15.6.5.4.3.2 Doses to a Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of the postulated LOCA have been conservatively analyzed, using assumptions and models described in previous sections.

The total-body dose due to immersion and the thyroid dose due to inhalation have been analyzed for the 0-2 hour dose at the exclusion area boundary and for the duration of the accident at the LPZ outer boundary. The results, with margin, are listed in Table 15.6-8. The resultant doses are within the guideline values of 10 CFR 100.

15.6.5.4.3.3 Doses to Control Room Personnel

Radiation doses to control room personnel following a postulated LOCA are based on the ventilation, cavity dilution, and dose model discussed in Section 15A.3.

Control room personnel are subject to a total-body dose due to immersion and a thyroid dose due to inhalation. These doses have been analyzed, and are provided in Table 15.6-8. The listed doses, with margin, are within the limits established by GDC-19.

15.6.6 A NUMBER OF BWR TRANSIENTS

This section is not applicable to the Callaway Plant.

15.6.7 REFERENCES

[Insert Section 15.6.5.5 from next page]

1. Huegel, D. S., et. al., “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” WCAP-14882-P-A, April 1999.

2. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure", WCAP-11397-P-A, April 1989.

3. SGTR Analysis letters SLNRC 86-01 (1-8-86), SLNRC 86-03 (2-11-86) SLNRC 86-05 (4-1-86), SLNRC 86-08 (9-4-86), ULNRC-1442 (2-3-87), ULNRC-1518 (5-27-87), ULNRC-1849 (10-21-88), ULNRC-2145 (1-29-90), and the NRC SER dated 8-6-90.

4. WCAP-16140, “Callaway Replacement Steam Generator Program NSSS Engineering Report,” June 2004.

5. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," 10 CFR 50.46, and, “ECCS Evaluation Models,” Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 4, 1974.

[INSERT TEXT]

15.6.5.5 Risk-Informed Assessment of Debris on Recirculation Sump Strainers

The licensing basis assessment for ECCS and containment heat removal with regard to effects of LOCA-generated debris on the containment recirculation sump strainers, as described in GL-2004-02, is a combined deterministic and risk-informed analysis that is described in more detail in Appendix 6.3A. The analysis examines the effects of debris on the strainers that support the ECCS and CSS core cooling and containment heat removal functions. The risk-informed analysis shows that there is a high confidence that the ECCS and CSS can perform their design basis functions based on plant-specific prototypical testing using deterministic assumptions that provide safety margin and defense-in-depth. The analysis also shows that the risk from breaks that could generate debris that is not bounded by the testing is very small in accordance with the criteria of Regulatory Guide 1.174.

The Callaway Risk-over-Deterministic (RoverD) methodology described in Appendix 6.3A was used to evaluate the effects of debris. RoverD relegates break sizes that generate and transport debris that is not bounded by plant-specific deterministic testing to failure (i.e., core damage). It then applies the NUREG-1829 (Reference 27) pipe break frequency for the smallest unbounded breaks to determine the increase in core damage frequency. The increase is compared to the criteria in Regulatory Guide 1.174 and is very small, as defined by Regulatory Guide 1.174. Exemptions to 10 CFR 50.46(a)(1), GDC 35, GDC 38 and GDC 41 have been approved to allow application of the risk-informed analysis instead of the deterministic methods required by GDCs. These exemptions apply to the scope of breaks that generate and transport debris not bounded by the plant-specific deterministic testing.

Engineering analysis in accordance with WCAP-16406 (Reference 28) and WCAP-17788 (Reference 29) shows that such debris will not affect downstream components external or internal to the reactor vessel, respectively.

The risk-informed analysis does not replace the ECCS evaluation methodology in the preceding FSAR Chapter 15.6 sections, which applies only through the LOCA re-flood phase. The Chapter 15.6 ECCS evaluation methodology is not used for the assessment of long-term cooling required by the risk-informed assessment of debris effects.

Loop and COSI Condensation Model", WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary), WCAP-10081-NP-Addendum 2, Revision 1 (Non-Proprietary), July 1997.

18. Deleted.

19. Letter from W. J. Johnson (Westinghouse) to T. E. Murley (NRC), "10CFR50.46 Annual Notification for 1989 of Modifications in the Westinghouse ECCS Evaluation Models", NS-NRC-89-3463, October 5, 1989.

20. Letter from W. J. Johnson (Westinghouse) to T. E. Murley (NRC), "Correction of Errors and Modifications to the NOTRUMP Code in the Westinghouse Small Break LOCA ECCS Evaluation Model Which are Potentially Significant", NS-NRC-89-3464, October 5, 1989.

21. "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A Revision 1 (Proprietary), and WCAP-15064-NP-A (Non-Proprietary), July 2000.

22. "Westinghouse Emergency Core Cooling System Evaluation Model - Sensitivity Studies", WCAP-8341 (Proprietary) and WCAP-8342 (Non-Proprietary), July 1974.

23. "Westinghouse Emergency Core Cooling System - Plant Sensitivity Studies," WCAP-8340 (Proprietary) and WCAP-8356 (Non-Proprietary), July 1974.

24. "Westinghouse ECCS – Four Loop Plant (17x17) Sensitivity Studies", WCAP-8565-P-A (Proprietary) and WCAP-8566-A (Non-Proprietary), July 1975.

25. NRC Information Notice 91-56, Potential Radioactive Leakage to Tank Vented to Atmosphere.

26. "Incorporation of the LOCBART Transient Extension Method into the 1981 Westinghouse Large Break LOCA Evaluation Model with BASH (BASH-EM)," WCAP-10266-P-A, Revision 2, Addendum 3 (Proprietary) and WCAP-11524-A, Revision 2, Addendum 3 (Non-Proprietary), December 2002. The December 2002 version was accepted by the NRC for RSG License Amendment 168 (NRC Safety Evaluation Reference 4.5).

27. ~~Deleted.~~ NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," April 2008.

28. ~~Deleted.~~ WCAP-16406-P-A, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191 (PA-SEE-0195)," Rev. 0, August 2007.

29. ~~Deleted.~~ WCAP-17788-P, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," Rev. 0, July 2015.